

Initial implementation of the neutronics codes DRAGON and DONJON at the North-West University

KM THOKWANE

 orcid.org/0000-0003-2751-1524

Dissertation accepted for the degree *Magister Scientiae* in
Nuclear Engineering at the North-West University

Supervisor: Dr VV Naicker
Graduation: May 2020
Student number: 30082706

ACKNOWLEDGEMENTS

Ke reta wena Morena, Sebautu, Swika la Ntlha, Sephiri. Ntle le wena ga re kgone selo. Ka Wena ke kgonne go fihla mo keleng gona lehono. Kea go leboga, ke reta ina la 'go go tsohle o mphileng tsona.

To my family, I thank you for the unwavering love and support. To my parents Mashego and Legopa Thokwane, I owe my life to you both. To my wife Mathapelo, thank you for being a constant source of strength, and your patience through the long hours which I spent on my research. Go bana baka (Remo le nyaga), you are my source of pride and joy.

To my study leader Dr Vishana Naicker, I am most grateful for your relentless leadership, guidance, lessons, and patience throughout this research project. I have learned a lot from your insights and expertise.

I would like to thank the staff at the North-West University and the Unit for Energy and Technology Systems for all the administrative support and teaching.

The DST Chair in Nuclear Engineering is hereby acknowledged for the financial support of this research.

ABSTRACT

The neutronics research group at the North West University plans to use open source neutronics in some of the future research that will be conducted by the group. The open source neutronics codes DRAGON and DONJON were identified as potential options.

However, since DRAGON and DONJON were not used in any research conducted at the North West University, the executables of the codes were created using the source codes openly available, and fuel assembly and full core models were developed.

These models were based on the North Anna reactor. DRAGON was used in modelling and performing lattice neutronics calculations for the fuel assemblies of the North Anna reactor. DONJON was used in modelling and performing full core calculations for the North Anna reactor.

A verification procedure for both the compilation and linking procedure together with the model development was employed. This employed using the models as supplied by the developers and building several intermediate models, which finally led to the North Anna reactor models.

The multiplication constant and flux results from the lattice and full core calculations were analysed using known reactor physics principles from literature. It was found that the results were consistent with what can be expected according to the reactor physics principles.

Therefore, executable versions of DRAGON and DONJON are obtained for which the source codes are available, and fuel assembly and full core models were developed, suitable as base model candidates for the research plans for the neutronics research group at the North West University. It is noted that a considerable amount of work is still required. This includes the addition of burnable poisons, modelling of control rods, burn-up studies, including spacer grids in the models, and coupling the neutronics results with thermal-hydraulic calculations.

Key Terms

Neutronics, Neutron-transport, Open-source codes, DRAGON, DONJON, NEWT, PWR, Lattice codes, Nodal diffusion codes, North Anna

LIST OF ABBREVIATIONS

1D	One dimension in space
2D	Two dimensions in space
USA	United States of America
GW	Giga Watts
U.S.NRC	United States Nuclear Regulation Commission
IRP	Integrated Resource Plan
VEPCO	Virginia Electrical Power Company
LANL	Los Alamos National Laboratory
ORNL	Oak Ridge National Laboratory
MC	Monte Carlo

TABLE OF CONTENTS

ACKNOWLEDGEMENTS	I
ABSTRACT	II
LIST OF ABBREVIATIONS	III
CHAPTER 1 INTRODUCTION.....	1
1.1 Background	1
1.2 Problem Statement.....	3
1.3 Research Aim.....	3
1.4 Research Steps.....	4
1.5 Report Outline.....	4
CHAPTER 2 LITERATURE SURVEY AND REACTOR DESCRIPTION.....	5
2.1 DRAGON	5
2.1.1 Introduction.....	5
2.1.2 DRAGON Theory.....	7
2.1.3 Neutron Transport Theory and the Neutron Transport Equation	7
2.1.4 Solution of the Neutron Transport Equation in DRAGON.....	11
2.1.5 The Infinite Multiplication Constant	12
2.1.6 The Equivalence Requirement.....	12
2.2 DONJON.....	15
2.2.1 Introduction.....	15
2.2.2 DONJON Theory	16
2.2.3 Nodal Diffusion Theory and the Multigroup Diffusion Equation	16

2.2.4	Solution of the Multigroup Diffusion Equation	17
2.2.5	The Effective Multiplication Constant	17
2.3	North Anna Reactor Description	18
2.3.1	Introduction.....	18
2.3.2	Reactor Safety Design.....	18
2.3.3	Reactor Core	19
2.3.4	Fuel Assemblies	21
2.3.5	Fuel Cells	22
2.3.6	Guide Tubes and Control Rods	24
CHAPTER 3	METHODOLOGY	27
3.1	Compiling and linking the DRAGON and DONJON Source Codes	27
3.2	North Anna Fuel Assembly Models: DRAGON	27
3.2.1	DRAGON Input File Syntax	28
3.2.2	General Structure of a DRAGON Input File	28
3.2.3	The Reference Model: TCWU01 – Case 4.....	38
3.2.4	From TCWU01 – Case 4 to the North Anna Assembly Models	39
3.3	The North Anna Full Core Model: DONJON	46
3.3.1	DONJON Input File Syntax.....	46
3.3.2	General Structure of a DONJON Input File	47
3.3.3	The Reference Model: LMW43D.....	53
3.3.4	From the Reference Model to the Full Core Model	56
CHAPTER 4	RESULTS AND DISCUSSION.....	62

4.1	The DRAGON Results	62
4.1.1	Compiling and Linking the DRAGON Source Code	62
4.1.2	The DRAGON Reference Model: TCWU01-Test Case 4	62
4.1.3	Infinite Multiplication Constants for the DRAGON Models.....	63
4.1.4	The 2-Group Constants for the North Anna Assembly Models	69
4.2	The DONJON Results	72
4.2.1	Compiling and Linking the DONJON Source Code	72
4.2.2	The DONJON Reference Model: LMW43D.....	72
4.2.3	Effective Multiplication Constants for the DONJON Models	73
4.2.4	Full Core (DON_10) Flux and Power Results	76
4.2.4.1	The Neutron Flux Profiles	76
4.2.4.2	The Power Profiles	84
CHAPTER 5 CONCLUSIONS AND RECOMMENDATIONS		89
5.1	Conclusions	89
5.2	Recommendations.....	89
BIBLIOGRAPHY		90

LIST OF TABLES

Table 2-1: DRAGON Modules and their Functions	6
Table 2-2: DONJON Modules and their Functions.....	16
Table 2-3: Properties of the North Anna Reactor Core.....	21
Table 2-4: Properties of the North Anna Fuel Assemblies.....	23
Table 2-5: Material Properties of the North Anna Fuel Cells.....	24
Table 2-6: Temperatures of the Fuel Cell Materials.....	25
Table 2-7: Material Properties of the Guide Tubes and Control Rods.....	27
Table 2-8: Guide Tube and Control Rod Temperatures.....	27
Table 3-1: Material Properties of TCWU01 - Test Case 4.....	40
Table 3-2: Dimensions of TCWU01 - Test Case 4.....	40
Table 3-3: Development of the North Anna Assembly Models.....	42
Table 3-4: LMW43D Dimensions and Mesh.....	56
Table 3-5: Development of the North Anna Full Core Model.....	59
Table 4-1: Comparison of the k_{∞} Values for TCWU01 - Test Case 4.....	65
Table 4-2: k_{∞} Values for the DRAGON Models.....	66
Table 4-3: The 2-group Constants for the Final Fuel Assembly Models.....	71
Table 4-4: k_{eff} Values for the Reference Model LMW43D.....	74
Table 4-5: k_{eff} Values for the DONJON Models.....	75
Table 4-6: Radial Neutron Flux along the Middle of Planes 1 and 8.....	80
Table 4-7: Axial Neutron Flux along the Center and Near the Core's Edge.....	85
Table 4-8: Radial Power along the Middle of Planes 1 and 8.....	88

Table 4-9: Axial Power along the Center and Near the Core's Edge.....	90
---	----

LIST OF FIGURES

Figure 2-0: Neutrons in differential volume dr at position r moving within cone $d\Omega$ about direction Ω	8
Figure 2-1: North Anna Reactor Core Cross Sectional View.....	20
Figure 2-2: Cross Sectional View of a North Anna Fuel Assembly.....	22
Figure 2-3: Cross Sectional View of a North Anna Fuel Cell.....	23
Figure 2-4: Cross Sectional View Guide Tube with Control Rod Inserted.....	26
Figure 3-1: TCWU01 - Test Case 4.....	41
Figure 3-2: DRAG_0.....	44
Figure 3-3: DRAG_1.....	44
Figure 3-4: DRAG_2.....	45
Figure 3-5: DRAG_3 to DRAG_9.....	45
Figure 3-6: DRAG_10.....	46
Figure 3-7: DRAG_11.....	46
Figure 3-8: DRAG_12 to DRAG_15.....	47
Figure 3-9: LMW43D Plane 1.....	56
Figure 3-10: LMW43D Plane 2 to Plane 19.....	57
Figure 3-11: LMW43D Plane 20.....	57
Figure 3-12: DON_1.....	60
Figure 3-13: DON_2.....	60
Figure 3-14: DON_3 and DON_4.....	61
Figure 3-15: DON_5.....	61

Figure 3-16: DON_6 and DON_7.....	62
Figure 3-17: DON_8.....	62
Figure 3-18: DON_9.....	63
Figure 4-1: Representative Assemblies for the Radial Flux Profiles of Each Plane.....	79
Figure 4-2: The 2-group Radial Flux Profiles for Planes 1 and 8.....	81
Figure 4-3: Plane 1 and 8 Flux Profile Decomposed into 2.1w/o and 2.6w/o Profiles.....	83
Figure 4-4: The 2-group Axial Flux Profiles along the Center and Edge of the Core.....	86
Figure 4-5: The 2-group Radial Power Profiles for Planes 1 and 8.....	89
Figure 4-6: The 2-group Axial Power Profiles along the Core's Center and Edge.....	91

CHAPTER 1 INTRODUCTION

1.1 Background

The South African government developed the Integrated Resource Plan 2010-2030 to address the electricity needs of the country in the period from 2010 to 2030. The objectives of the Integrated Resource Plan are to determine South Africa's projected electricity demand from 2010 to 2030, and how to best meet that demand (DoE, 2019).

In the 2010 iteration of the Integrated Resource Plan, it was determined that an additional 53 GW of electrical capacity would be required to meet the projected electricity demand by 2030 (DoE, 2010). Of that 53 GW, 9.6 GW was allocated to new nuclear power as a source of baseload electricity (DoE, 2010).

However, in the 2018 iteration of the Integrated Resource Plan, it was shown that the projected electricity demand as reported in the 2010 iteration was not realized (DoE, 2018). It was also reported that the proposed new nuclear power must be left out of the energy mix, and that it should be reconsidered after 2030 (DoE, 2018). The 2018 iteration was published to the public for comments. The final version, which was published in October of 2019, reports that nuclear power will be included in the energy mix, even before 2030 (DoE, 2019).

In support of the impending new nuclear build in South Africa, it is important to have local nuclear engineering knowledge and skills (DoE, 2010). One of the important nuclear engineering skills to localise is nuclear reactor analysis.

Nuclear reactor analysis is the characterization of a nuclear reactor in terms of its neutronic properties (Dictionary, 2018). Neutronic properties include the distribution of neutrons in a nuclear reactor, the temporal-spatial changes of this distribution and the interaction of the neutrons with materials within the reactor. The neutron flux represents this distribution. It is defined as the product of neutron volume density (neutrons/cm³) with the neutron speed (cm/s (U.S.NRC, n.d.) at some space-time coordinate in the reactor.

For a given reactor system, the flux can in principle be determined by solving the neutron transport equation (see section 2.1). Being an integro-differential equation with seven independent variables (three spatial, two angular, one energy and one time), the neutron transport equation is typically impossible to solve exactly (Encyclopedia, 2017) except in geometrically simple systems. Even in these cases, the solution features some special functions which are difficult to handle (Liou, 1973).

In the case of realistic reactor geometries, no known analytical methods for solving the Neutron Transport Equation exist. Instead, approximate solutions are acquired using numerical techniques. Two of these techniques are the Discrete Ordinates method and the Monte Carlo method. The Discrete Ordinates method involves discretizing the neutron transport equation input variables in order to convert it into a system of ordinary differential equations (Stefan, 1998) which can then be solved using linear algebra techniques, such as Gauss-Jordan elimination (Roberts, 2010). On the other hand, the Monte Carlo approach uses probability concepts to simulate the neutron random walk in the reactor materials (Lux, 2018).

In both cases, the calculations involved demand considerable computational resources (Lux, 2018). With the discrete ordinates method, the resulting equations are often too numerous to solve manually (Encyclopedia, 2017). With the Monte Carlo method, the law of large numbers requires many random walk simulations before any insightful results can be obtained (Britannica, 2019). Therefore, computers are necessary to obtain a solution in both methods. The computer software used for this purpose are called neutronics codes.

Using neutronics codes, a model of a reactor system can be developed by specifying the reactor's dimensions, material composition and temperatures. By solving the neutron transport equation for that model using the methods outlined above, the neutronic behaviour of the reactor system can be predicted. These predictions are useful for safety analysis, criticality calculations and reactor physics research in general. A level of uncertainty is usually associated with the results obtained so that they can be used with a known level of confidence.

The ongoing challenge in reactor neutronics research is the development of neutronics codes which can predict the neutronics of a reactor with the highest possible accuracy. In response to this, a variety of neutronics codes have been developed over the years. Various institutions with a focus on this research have also emerged over the years, including academic institutions. The North-West University (NWU) is one such institution.

The neutronics group at the NWU actively participates in neutronics research. Both Monte Carlo and Discrete Ordinates codes have been used in the past (MCNP (LANL, 2019) and NEWT of SCALE (ORNL, 2019) respectively). An in-house code, NWURCS (Naicker, et al., 2015) was developed by the group for generating MCNP and NEWT input files.

Some neutronics codes are published without their source code, e.g. MCNP, SCALE. They are called closed source codes. With closed source codes, the end-users must apply for special access and satisfy often intensive screening in order to access the source code. Closed source

codes are usually commercialized. On the other hand, some codes are published along with their source code, commonly free of charge.

The neutronics codes used by the NWU neutronics group in the past are closed source. The neutronics group's vision is to use open source neutronics codes in some of their future research projects. One advantage of using open source codes is that the usage of the codes become less of "black box" operations. For example, when the understanding of a given algorithm is difficult, the source code dealing with the particular algorithm can be studied in order to gain further insight. In line with this vision, this research project is designed to explore two such neutronics codes, namely DRAGON and DONJON.

In this research project, the open source neutronics codes DRAGON and DONJON are tested by modelling a simple steady state of the North Anna reactor core. The North Anna reactor is a Pressurized Water Reactor designed by Westinghouse and installed at the North Anna Nuclear Power Station in Louisa County, Virginia, USA (VEPCO, 2009).

The North Anna reactor was chosen for testing DRAGON and DONJON because it is similar in design to the Koeberg reactor. Therefore, the findings of this research project are relevant in the South African context.

1.2 Problem Statement

In the past the neutronics codes used by the NWU neutronics group in research projects were closed source. Although they served the group's calculation needs, being closed source meant that access to their source code was restricted. Intense screening was required for access to be granted, which was time consuming and costly. Having access to the source code would enable the group to understand explicitly how the codes implemented the various theoretical calculation algorithms when performing neutronics calculations. Such observations would assist the group's methodologies development research and contribute to a better and deeper understanding of the results obtained when using those codes.

1.3 Research Aim

The aim of this research project is to compile locally and to apply the open source neutronics codes DRAGON and DONJON to the North Anna reactor.

1.4 Research Objectives and planned Methodology

The research objectives are to obtain the compiled source code of both DRAGON and DONJON, as well as to produce the assembly and full core models of the North Anna reactor. The planned methodology to realise these objectives is as follows::

- Acquire the DRAGON and DONJON source codes.
- Write Bash scripts to compile and link those source codes.
- Write DRAGON and DONJON input files for the North Anna reactor.
- Execute the DRAGON and DONJON input files.
- Analyse the DRAGON and DONJON results obtained.

The results from applying the compiled source codes on the North Anna reactor will be analysed to check if they are consistent with reactor physics theory. If they are, then the compiled source code and the reactor models can be considered to be acceptable for use in future work.

1.5 Report Outline

This research report is structured as follows:

Chapter 1 discusses some of the challenges that the energy landscape of South Africa faces, the government's plans in response to those challenges, and the role that nuclear power may play in those plans.

Chapter 2 presents a literature survey of the physics theories which are implemented in DRAGON and DONJON. It also includes details of the North Anna reactor such as safety design, core layout and materials.

Chapter 3 explores the syntax and structure of input files for DRAGON and DONJON. It also details the methodology followed in acquiring, compiling and linking the DRAGON and DONJON source codes, as well as developing the assembly and full core models of the North Anna reactor.

Chapter 4 presents the results obtained by following the methodology described in *Chapter 3*, as well as a discussion of these results.

Chapter 5 gives conclusions drawn from *Chapter 4*, and recommendations for future work which can be done to extend the scope of this research project.

CHAPTER 2 LITERATURE SURVEY AND REACTOR DESCRIPTION

2.1 DRAGON

2.1.1 Introduction

DRAGON is a modular deterministic neutronics lattice code developed at Ecole Polytechnique de Montreal in Canada (Marleau, et al., 2014). As a modular code, it is a collection of independent modules which are connected using the GAN generalized driver (Hebert, 2000) so that communication from one module to another is through well-defined data structures (Marleau, et al., 2014).

Table 2-1 below summarizes the DRAGON modules and their functions. Depending on what a user needs to do in DRAGON, different modules can be called while some can be left out of the calculation scheme. For instance, to produce a digital image of a given geometry, only the geometry (GEO:), tracking (ASM:, EXCELL:) and graphics(PSP:) modules are required.

Table 2-1: DRAGON Modules and their Functions (Marleau, et al., 2014)

Modules	Functions
MAC: and LIB:	Read material properties such as cross-sections, temperatures and densities.
GEO:	Reads geometry properties such dimensions and boundary conditions.
EXCELT:, NXT:, SYBILT: and JPMT:	Responsible for performing tracking calculations on a geometry defined in the module GEO:.
SHI:	Responsible for performing resonance self-shielding calculations for the materials defined in the MAC: and LIB: modules.
ASM: and EXCELL:	Prepare collision probability matrices which are required for flux solution calculations
FLU:, MOCC: and MCU:	Perform flux calculations using the matrices prepared by the ASM: and EXCELL modules.
EDI:	Performs group collapse and spatial homogenization calculations. It also performs equivalence calculations to check if the reaction rate is conserved during the group collapse and homogenization calculations.
EVO:	Performs burn-up calculations
CPO: and CFC:	Captures the group collapsed, homogenized cross-sections prepared by the EDI: module, and produces a binary output file which can be used by a full-core code such as DONJON.
PSP:	Generates a digital image of a geometry defined in the GEO: module.

2.1.2 DRAGON Theory

DRAGON is a lattice code. Therefore, it performs neutron transport calculations on fuel assembly models to determine their flux and infinite multiplication factor (Marleau, et al., 2014).

In DRAGON, a fuel assembly model is developed by specifying the assembly's geometry and its material properties such as temperatures and compositions. To simplify the modelling process, the fuel assembly is assumed to be surrounded by an infinite array of other fuel assemblies (Stacey, 2001). This assumption is implemented in DRAGON by specifying reflective boundary conditions for the assembly model (Marleau, et al., 2014).

DRAGON also requires nuclear data libraries. These nuclear data libraries contain microscopic cross-section data prepared using data obtained from nuclear experiments which are conducted over many collision energies. For lattice codes, the number of collision energies must be reduced to 69-400. This is necessary to reduce the computational demands of lattice calculations. To that end, nuclear data processing codes such as NJOY (NJOY, 2016) must be first used to create collapsed data sets which are then used by the lattice codes such as DRAGON.

2.1.3 Neutron Transport Theory and the Neutron Transport Equation

Neutron transport theory is the study and characterization of neutron motion and interactions in different media (Stacey, 2001). It falls under the broader theory of linear transport theory (Case & Zweifel, 1967). Its mathematical formulation stems from the Boltzmann transport equation (Case & Zweifel, 1967) which was constructed by Ludwig Boltzmann in 1872 to model the non-equilibrium behaviour of a thermodynamic system.

The mathematical formulation of neutron transport theory is in the form of the neutron transport equation. As expressed in equation 2-1 below (Stacey, 2001), the neutron transport equation states that the rate of change of the neutron density in the system is equal to the total rate of neutron production plus the total rate of neutron loss in the system. Each term in the equation corresponds to either a decrease or increase of neutrons in the system. Consider Figure 2.0 below

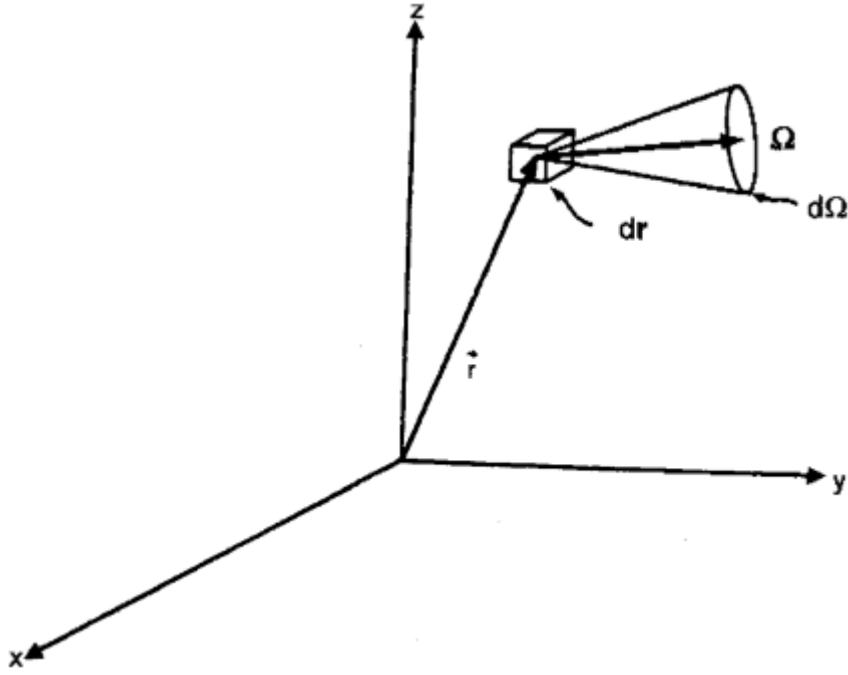


FIGURE 2-0: Neutrons in differential volume dr at position $\mathbf{r}$ moving within cone $d\Omega$ about direction Ω (Stacey, 2001).

$$\begin{aligned}
 \frac{\partial N}{\partial t}(\mathbf{r}, \Omega, t) dr d\Omega = & v(N(\mathbf{r}, \Omega, t) - N(\mathbf{r} + \Omega dl, \Omega, t)) dA d\Omega \\
 & + \int_0^{4\pi} d\Omega' \Sigma_s(\mathbf{r}, \Omega' \rightarrow \Omega) v N(\mathbf{r}, \Omega', t) dr d\Omega \\
 & + \frac{1}{4\pi} \int_0^{\pi} d\Omega' v \Sigma_f(\mathbf{r}) v N(\mathbf{r}, \Omega', t) dr d\Omega \\
 & + S_{ex}(\mathbf{r}, \Omega) dr d\Omega - (\Sigma_a(\mathbf{r}) + \Sigma_s(\mathbf{r})) v N(\mathbf{r}, \Omega, t) dr d\Omega
 \end{aligned}
 \tag{2-1}$$

In this equation

$\mathbf{r}$ is the position vector (x, y, z);

Ω is the direction vector;

t is time;

v is the neutron speed;

$N(\mathbf{r}, \boldsymbol{\Omega}, t) d\mathbf{r}d\boldsymbol{\Omega}$ is the number of neutrons within a differential volume $d\mathbf{r}$ traveling within differential direction $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$;

$\frac{\partial N}{\partial t}(\mathbf{r}, \boldsymbol{\Omega}, t) d\mathbf{r}d\boldsymbol{\Omega}$ is the rate of change of number of neutrons within a differential volume $d\mathbf{r}$ traveling within differential direction $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$;

Σ_s is the macroscopic neutron scattering cross-section;

Σ_a is the macroscopic neutron absorption cross-section;

Σ_f is the macroscopic neutron fission cross-section;

$\int_0^{4\pi} d\boldsymbol{\Omega}' \Sigma_s(\mathbf{r}, \boldsymbol{\Omega}' \rightarrow \boldsymbol{\Omega}) v N(\mathbf{r}, \boldsymbol{\Omega}', t) d\mathbf{r}d\boldsymbol{\Omega}$ is the rate at which neutrons which were travelling in the directions $\boldsymbol{\Omega}'$ are scattered into the differential volume $d\mathbf{r}$ within $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$;

$\frac{1}{4\pi} \int_0^\pi d\boldsymbol{\Omega}' v \Sigma_f(\mathbf{r}) v N(\mathbf{r}, \boldsymbol{\Omega}', t) d\mathbf{r}d\boldsymbol{\Omega}$ is the rate at which neutrons are produced from fission reactions within differential volume $d\mathbf{r}$ and direction $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$;

$S_{ex}(\mathbf{r}, \boldsymbol{\Omega}) d\mathbf{r}d\boldsymbol{\Omega}$ is the rate at which neutrons are introduced into differential volume $d\mathbf{r}$ and direction $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$ from external sources;

$(\Sigma_a(\mathbf{r}) + \Sigma_s(\mathbf{r})) v N(\mathbf{r}, \boldsymbol{\Omega}, t) d\mathbf{r}d\boldsymbol{\Omega}$ is the rate at which neutrons in differential volume $d\mathbf{r}$ and direction $d\boldsymbol{\Omega}$ of $\boldsymbol{\Omega}$ are absorbed or scattered into directions $\boldsymbol{\Omega}'$.

The neutron transport equation was formulated with the following assumptions (Lewis & Miller, 1985):

- Neutrons are treated as point particles;
- Neutron to neutron collisions are assumed to be negligible;
- Collisions are assumed to occur instantaneously;
- The neutrons are assumed to only travel in straight lines from one collision to the next;
- Material properties are assumed to remain the same until they are explicitly changed; and
- Only the mean value of the neutron density distribution is considered.

The neutron transport equation can be written in terms of the neutron flux $\psi(\mathbf{r}, \boldsymbol{\Omega}, t)$ as shown in equation 2-2, by making the following substitutions (Stacey, 2001): $\psi(\mathbf{r}, \boldsymbol{\Omega}, t) = vN(\mathbf{r}, \boldsymbol{\Omega}', t)$, $\Sigma_t = \Sigma_a + \Sigma_s$, and $\Sigma_s(\mathbf{r}, \boldsymbol{\Omega} \rightarrow \boldsymbol{\Omega}') = \frac{1}{2\pi} \Sigma_s(\mathbf{r}, \mu_0)$, where $\mu_0 = \boldsymbol{\Omega}' \cdot \boldsymbol{\Omega}$.

$$\begin{aligned} & \frac{1}{v} \frac{\partial \psi}{\partial t}(\mathbf{r}, \boldsymbol{\Omega}, t) + \boldsymbol{\Omega} \cdot \nabla \psi(\mathbf{r}, \boldsymbol{\Omega}, t) + \Sigma_t(\mathbf{r}) \psi(\mathbf{r}, \boldsymbol{\Omega}, t) \\ &= \int_{-1}^1 d\mu_0 \Sigma_s(\mathbf{r}, \mu_0) \psi(\mathbf{r}, \boldsymbol{\Omega}', t) + \frac{1}{4\pi} \int_0^{4\pi} d\boldsymbol{\Omega}' v \Sigma_f(\mathbf{r}) \psi(\mathbf{r}, \boldsymbol{\Omega}', t) + S_{ex}(\mathbf{r}, \boldsymbol{\Omega}) \end{aligned} \quad (2-2)$$

A scaling factor of $\frac{1}{k}$ can be introduced in equation 2-2 to ensure that the rate of neutron production is equal to the rate of neutron loss. In that case, the neutron flux in equation 2-2 would not change over time. Therefore, the time derivative of the flux in equation 2-2 would vanish. The resulting equation is the steady-state neutron transport equation. For $k = 1$, the steady state neutron transport equation can be written in the following form (Stacey, 2001):

$$\frac{d}{dR} \psi(\mathbf{r}, \boldsymbol{\Omega}) d\mathbf{r} d\boldsymbol{\Omega} + \Sigma_t(\mathbf{r}) \psi(\mathbf{r}, \boldsymbol{\Omega}) d\mathbf{r} d\boldsymbol{\Omega} = S(\mathbf{r}, \boldsymbol{\Omega}) d\mathbf{r} d\boldsymbol{\Omega} \quad (2-3)$$

Where:

$$\frac{d}{dR} = \boldsymbol{\Omega} \cdot \nabla$$

dR is a differential length along the direction $\boldsymbol{\Omega}$

$$S(\mathbf{r}, \boldsymbol{\Omega}) = \frac{1}{4\pi} \int_0^{4\pi} d\boldsymbol{\Omega}' v \Sigma_f(\mathbf{r}) \psi(\mathbf{r}, \boldsymbol{\Omega}') + S_{ex}(\mathbf{r}, \boldsymbol{\Omega})$$

The steady state neutron transport equation (2-3) may be integrated from $\mathbf{r}_0$ to $\mathbf{r}$ along the direction $\boldsymbol{\Omega}$. The resulting equation is the integral steady-state neutron transport equation, shown in equation 2-4 below (Stacey, 2001):

$$\psi(\mathbf{r}, \boldsymbol{\Omega}) d\mathbf{r} = e^{-\alpha(\mathbf{r}_0, \mathbf{r})} \psi(\mathbf{r}_0, \boldsymbol{\Omega}) d\mathbf{r}_0 + \frac{1}{k} \int_{\mathbf{r}_0}^{\mathbf{r}} e^{-\alpha(\mathbf{r}', \mathbf{r})} S(\mathbf{r}', \boldsymbol{\Omega}) d\mathbf{r}' \quad (2-4)$$

In equation 2-4, the quantity $\alpha(\mathbf{r}', \mathbf{r})$ is the optical path length between $\mathbf{r}'$ and $\mathbf{r}$ along the direction $\boldsymbol{\Omega}$, given by (Stacey, 2001):

$$\alpha(\mathbf{r}', \mathbf{r}) = \left| \int_{\mathbf{r}'}^{\mathbf{r}} \Sigma_t(R) dR \right| \quad (2-5)$$

For a 3D model, the geometry may be discretized into finite volume elements V_i and it may be assumed that within each volume element the flux is constant and the cross-sections are uniform (Stacey, 2001). Using this assumption and equation 2-4, the total neutron flux in each volume element V_i takes the following form (Stacey, 2001):

$$\phi_i = \sum_j T^{j \rightarrow i} [(\Sigma_{sj} + \nu \Sigma_{fj}) \phi_j + S_{0j}] \quad (2-6)$$

Where the quantity $T^{j \rightarrow i}$ is the transmission probability of neutrons from volume elements V_j into the volume element V_i . For each volume element V_j , $T^{j \rightarrow i}$ is given by (Stacey, 2001):

$$T^{j \rightarrow i} = \frac{1}{V_j} \int d\mathbf{r}_i \int d\mathbf{r}_j \frac{e^{-\alpha(\mathbf{r}', \mathbf{r})}}{4\pi|\mathbf{r}_j - \mathbf{r}_i|^2} \quad (2-7)$$

Multiplication of equation 2-7 by $\Sigma_{ti}\Sigma_{tj}V_i$, the following expression is obtained (Stacey, 2001):

$$P^{ji} = \Sigma_{ti}\Sigma_{tj}V_i T^{j \rightarrow i} \quad (2-8)$$

Where P^{ji} is called the collision probability of neutrons. It is the probability that neutrons which originate from volume elements V_j will experience their first collision in volume element V_i . In terms of the collision probabilities, equation 2-6 is as follows:

$$\phi_i = \sum_j P^{ji} \frac{[(\Sigma_{sj} + \nu \Sigma_{fj}) \phi_j + S_{0j}]}{\Sigma_{tj}} \quad (2-6)$$

Equations 2-6 and 2-8 are then solved in DRAGON as described next in Section 2.1.4.

2.1.4 Solution of the Neutron Transport Equation in DRAGON

For a given assembly model, the DRAGON tracking modules analyze the geometry which is specified in the GEO: module. The main tasks of the tracking modules are to perform surface area calculations and to generate the integration lines for the geometry (Marleau, et al., 2014). The surface area calculations are for discretizing the geometry into finite areas A_i (Marleau, 2001).

The collision probability modules then use the tracking information together with the material properties to compute collision probabilities (equation 2-8) for the model (Marleau, 2001). In addition to computing the collision probability matrices, the collision probability modules also prepare the linear system of collision probability equations (equation 2-6) (Marleau, et al., 2014).

The system of Collision Probability equations are then solved by the flux solution module FLU:. The solutions of the system of equations are the assembly flux and infinite multiplication constant.

2.1.5 The Infinite Multiplication Constant

The infinite multiplication factor is defined as the ratio of the number of fission reactions in one generation to the number of fission reactions in the preceding generation in the absence of neutron leakage (Lamarsh & Barrata, 2001).

It is obtained in DRAGON when the system of collision probability equations is solved.

2.1.6 The Equivalence Requirement

While lattice calculations consider heterogeneous assemblies where cross-sections for the various materials and fluxes are specified as fine-group values, full core calculations consider homogenous assemblies in few-groups to avoid impractical computational requirements. To preserve the physics of a heterogeneous assembly, the properties of a corresponding homogeneous few-group assembly are required to yield the same total rate of reaction as with the heterogenous fine-group assembly. This is the equivalence requirement, and the calculations are said to be equivalent if this is satisfied.

Given a heterogeneous cell of R regions with volumes V_r ($r = 1, 2, \dots, R$) of which fluxes and macroscopic cross-sections are specified in G fine groups ($\phi_r^g, \Sigma_r^g, g = 1, 2, \dots, G$), then the total reaction rate in the region r and group g is given by equation 2-9 (Lamarsh & Barrata, 2001). The total reaction-rate in the cell is then given by summing equation 2-9 over the cell's regions as in equation 2-10. Similarly, if a fuel assembly is made up of C heterogeneous cells each with R_c ($c = 1, 2, \dots, C$) regions, the total assembly reaction-rate would be given by the sum of the individual cell reaction-rates as in equation 2-11.

$$RR_r^g = V_r \Sigma_r^g \phi_r^g \quad (2-9)$$

$$RR_{c(het)}^g = \sum_{r=1}^R (V_r \Sigma_r^g \phi_r^g) \quad (2-10)$$

$$RR_{ass(het)}^g = \sum_{c=1}^C (\sum_{r_c=1}^{R_c} (V_{r_c} \Sigma_{r_c}^g \phi_{r_c}^g)) \quad (2-11)$$

A homogeneous assembly which is equivalent to the heterogeneous one described above must have macroscopic cross-section and flux values (called effective cross-section and flux) which yield the same result as equation 2-11. To deduce those cross-section and flux values, spatial homogenization and group-collapse calculations must be done.

With spatial homogenization, known properties of a heterogeneous system are used to determine the properties of an equivalent homogeneous system (Stacey, 2001). For the heterogeneous cell described above, the effective macroscopic cross-section is defined as in equation 2-12. It is the sum of the volume-flux weighted macroscopic cross-sections taken over all the regions, divided by the sum of the volume-flux weights taken over all the regions. The effective flux is defined as in equation 2-13. It is given by the sum of the volume weighted fluxes of the various regions, divided by the total volume of the cell.

The volume weight accounts for the effect of each region's size on its contribution to the effective cross-section, while the flux weight accounts for importance of the neutrons in terms of spatial distribution and energy group.

$$\Sigma_{c(hom)}^g = \frac{\sum_{r=1}^R (V_r \Sigma_r^g \phi_r^g)}{\sum_{r=1}^R (V_r \phi_r^g)} \quad (2-12)$$

$$\phi_{c(hom)}^g = \frac{\sum_{r=1}^R (V_r \phi_r^g)}{\sum_{r=1}^R (V_r)} \quad (2-13)$$

For the heterogeneous assembly of C heterogeneous cells described above, the effective assembly macroscopic cross-section and flux are obtained by taking the summations in equations 2-12 and 2-13 respectively over the cells of the assembly. The effective assembly macroscopic cross-section and flux are expressed in equations 2-14 and 2-15 respectively.

$$\Sigma_{ass(hom)}^g = \frac{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} (V_{rc} \Sigma_{rc}^g \phi_{rc}^g))}{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} (V_{rc} \phi_{rc}^g))} \quad (2-14)$$

$$\phi_{ass(hom)}^g = \frac{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} (V_{rc} \phi_{rc}^g))}{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} (V_{rc}))} \quad (2-15)$$

Equations 2-14 and 2-15 characterize the homogenized assembly. The reaction rate of this assembly, given by equation 2-16 below, must equal equation 2-11 as per equivalence requirement.

$$RR_{ass(hom)}^g = V_{ass} \Sigma_{ass(hom)}^g \phi_{ass(hom)}^g \quad (2-16)$$

Substituting equations 2-14 and 2-15 into equation 2-16:

$$RR_{ass(hom)}^g = V_{ass} \left(\frac{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} V_{r_c} \Sigma_{r_c}^g \phi_{r_c}^g)}{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} V_{r_c} \phi_{r_c}^g)} \right) \left(\frac{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} V_{r_c} \phi_{r_c}^g)}{\sum_{c=1}^C (\sum_{r_c=1}^{R_c} V_{r_c})} \right)$$

The denominator of the first fraction and the numerator of the second fraction are identical, thus they cancel out. Moreover, the denominator of the second term is the summation of volumes over the whole assembly, so it is equal to the assembly volume V_{ass} . Therefore, the assembly volume V_{ass} cancels the denominator of the second fraction to yield:

$$RR_{ass(hom)}^g = \sum_{c=1}^C \sum_{r_c=1}^{R_c} V_{r_c} \Sigma_{r_c}^g \phi_{r_c}^g \quad (2-17)$$

The reaction rate of the homogeneous assembly (equation 2-17) is therefore equal to the reaction rate of the heterogeneous assembly (equation 2-11). The two assemblies are therefore equivalent.

Therefore, the homogeneous assembly can be considered in a full core calculation in place of the heterogeneous one. However, the homogeneous flux and cross-section are still fine-group values. Group-collapse must be applied to reduce the number of groups.

With group-collapse, the number of energy groups in which the cross-section and flux values are specified are reduced by merging some of the fine-groups into broad-groups. For a two-group homogeneous assembly, the energy spectrum is divided into two broad intervals: *group 1* extends from the former $g = 1$ to some group h while *group 2* extends from $g = h + 1$ to the former $g = G$, where $h \in \{g \mid g=1, \dots, G\}$. The equivalence requirement also applies here. The total reaction rate of the fine-group assembly must be equal to the total reaction rate of the broad-group assembly.

2.2 DONJON

2.2.1 Introduction

DONJON is a modular deterministic neutronics full core code developed at Ecole Polytechnique de Montreal in Canada (Varin, et al., 2005). It is a collection of independent modules which are connected through the GAN generalized driver (Hebert, 2000). Communication between the modules is through well-defined data structures (Varin, et al., 2005).

Table 2-2 below summarizes the DONJON modules and their functions. Depending on a user's needs, different combinations of modules can be called while some can be left out of the calculation scheme. For example, the module REFUEL: is not called in a calculation which does not involve burn-up.

Table 2-2: DONJON Modules and their Functions (VEPCO, 2009)

Modules	Functions
MACD: and CRE:	Read material properties such as cross-sections, temperatures and densities.
GEOD:	Reads geometry properties such as dimensions and boundary conditions.
BIVACT: and TRIVAT:	Perform tracking calculations for a geometry defined in the GEOD: module.
BIVAA: and TRIVAA	Compute collision probability matrices
FLUD:	Performs flux calculations using the matrices from BIVAA: and TRIVAA:
REFUEL:	Performs burn-up calculations

2.2.2 DONJON Theory

DONJON is a full core neutronics code. It performs nodal diffusion calculations on full core models.

It takes as input the reactor database generated by DRAGON along with a full core geometric model to determine the neutronics for the core (Varin, et al., 2005). To take into account the finite size of the reactor core, DONJON full core models are assigned void boundary conditions.

2.2.3 Nodal Diffusion Theory and the Multigroup Diffusion Equation

For a given volume, the neutrons are assumed to be under diffusion-like motion characterized by Fick's law (Lamarsh & Barrata, 2001). Fick's law states that particles in a medium spontaneously spread from regions where they are in high concentration to regions where they are in low concentration (Lamarsh & Barrata, 2001). Its mathematical statement relates the nodal neutron flux ϕ_g to the neutron current density J_g of the considered volume of energy group g via the diffusion constant D_g , as shown in equation 2-18 below

$$J_g = -D^g \nabla \phi_g \quad (2-18)$$

Where

J_g is the neutron current density;

D^g is the diffusion coefficient;

ϕ_g is the neutron flux;

Using this relation of the flux with the current, the multigroup diffusion equation is stated as follows (Stacey, 2001):

$$\sum_{g' \rightarrow g}^G \Sigma_s^{g' \rightarrow g} \phi_{g'} + \frac{1}{k} \chi^g \sum_{g'=1}^G \nu \Sigma_f^{g'} \phi_{g'} = \Sigma_a^g \phi_g - \nabla \cdot D^g \nabla \phi_g \quad (2-19)$$

Where

g is the group index, where g is between 1 and some number G ;

k is the effective multiplication factor;

ν is the average number of neutrons released per fission reaction;

χ^g is the fraction of fission neutrons which are released with energy belonging to group g ;

ϕ_g is the neutron flux in group g ;

$\Sigma_f^{g'}$ is the macroscopic fission cross-section in group g' ;

Σ_a^g is the macroscopic absorption cross-section in group g ; and

$\Sigma_s^{g' \rightarrow g}$ is the macroscopic scattering cross-section from group g' to g .

In nodal diffusion theory, each fuel assembly in the core is considered as a node made of an effective material. Moreover, broad group (typically 2-group) libraries are used. Thus, the full core flux solution is determined as one value per fuel assembly in few energy groups ϕ_g .

2.2.4 Solution of the Multigroup Diffusion Equation

The full core geometry defined is analysed by the tracking modules to develop a finite element mesh for the geometry (Varin, et al., 2005). The tracking information is then used by the system modules to compute the system of equations for the core model (Varin, et al., 2005).

DONJON solves the system of equations using the Power Method (Varin, et al., 2005). Also, known as the Von Mises iteration, the power method is an iterative eigenvalue algorithm which solves for the optimum eigenvalue and eigen vector pair for a given matrix equation (Ipsen, 2009).

The solutions to the system of equations are the few-group flux and the effective multiplication constant for the full core models.

2.2.5 The Effective Multiplication Constant

The effective multiplication factor is defined as the ratio of the number of fission reactions in one generation to the number of fission reactions in the preceding generation when neutron leakage is considered (Lamarsh & Barrata, 2001). It is obtained in DONJON during the solution of the nodal diffusion equation.

2.3 North Anna Reactor Description

2.3.1 Introduction

The North Anna power station is a nuclear power plant located in Louisa County, Virginia, USA. It supplies about 1.79 GW of electric power per year, mainly to the Richmond area in Northern Virginia. The plant was constructed by Virginia Electric Power Company (VEPCO) together with Stone & Webster Engineering Corporation between 1970 and 1980 (VEPCO, 2009).

Two reactor units were constructed, with unit 1 commencing operation in 1978 and unit 2 in 1980. Each unit includes a Westinghouse pressurized light water reactor connected to a Westinghouse 3-loop nuclear steam supply system housed in a steel-lined, reinforced-concrete containment building (VEPCO, 2009).

2.3.2 Reactor Safety Design

The North Anna reactor was designed to adhere to safety standards as required by the United States Nuclear Regulation Commission (VEPCO, 2009). The safety measures which are in place to prevent or manage nuclear accidents for the North Anna reactor are discussed below.

- Fission Products Containment (VEPCO, 2009)

The fission products which accumulate when the fuel undergoes fission are contained by three independent barriers to minimize their probability of escaping into the environment. The first barrier is the fuel pin cladding which encloses the fuel pellets. The second barrier is the reactor pressure vessel, which encloses the fuel core itself. The third and outermost barrier is the containment building, which is made of reinforced concrete.

- Decay Heat Removal (VEPCO, 2009)

The North Anna reactor is connected to a 3 loop Westinghouse steam supply system to remove the heat generated from the core during normal operation. During shut down or accident scenarios, an auxiliary coolant system is available to supply the core with coolant for removing the decay heat.

- Chemical Stability (VEPCO, 2009)

The helium gas used in the plenum of the fuel rods is chemically inert. Therefore, it does not react with the fuel pellets, fission gas or the cladding of the fuel rods.

The uranium dioxide fuel is resistant to corrosion by the water moderator. Thus, in the event of cladding defects, fuel deterioration is limited.

The Zircaloy-4 cladding is also corrosion resistant to the moderator.

- Mechanical Stability (VEPCO, 2009)

The uranium dioxide fuel remains intact for temperatures of up to about 3000 K. The operating conditions of the reactor are kept below this temperature to avoid melting of the fuel.

The fuel rod cladding was designed to withstand larger mechanical stresses than the mechanical stress which is present in the core.

The reactor pressure vessel was designed to withstand the high pressures in the reactor core.

2.3.3 Reactor Core

The North Anna reactor core is made up of 193 uranium dioxide fuel assemblies of three different enrichments: 3.1 w/o, 2.6 w/o and 2.1 w/o, where w/o means weight percent (Lamarsh, 2001) of U-235. The 3.1 w/o assemblies are placed at the periphery of the core. The 2.6 w/o and 2.1 w/o assemblies are arranged in a selected pattern in the interior of the core. A cross sectional view of the North Anna core is shown in Figure 2-1 below.

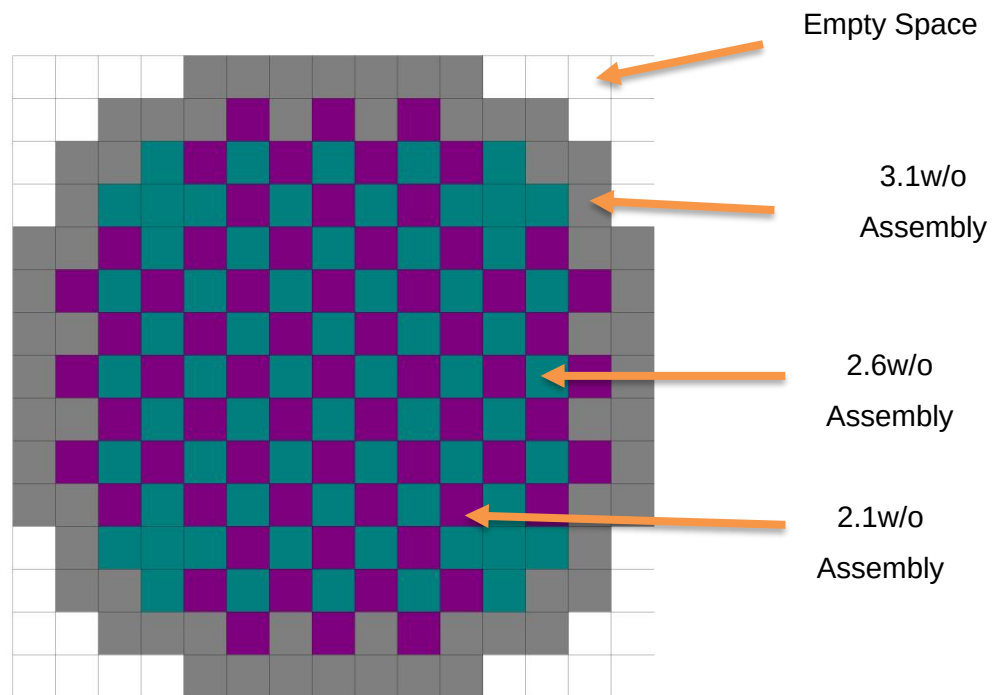


Figure 2-1: North Anna Reactor Cross Sectional View

Table 2-3 below lists the major design parameters of the North Anna reactor core.

Table 2-3: Properties of the North Anna Reactor Core (VEPCO, 2009)

Properties	Values
Thermal Power	2775 MWth
Pressure	15.513 MPa
Mass Flow Rate	3282.1 kg/m ² s
Inlet Temperature	559.15 K
Outlet Temperature	596.37 K
Active Core Height	365.998 cm
Equivalent Core Diameter	304.038 cm
No. of Fuel Assemblies	157
Fuel Enrichments	3.1w/o, 2.6w/o and 2.1w/o

2.3.4 Fuel Assemblies

The North Anna reactor fuel assemblies are 17 x 17 arrays. In each assembly there are 264 fuel rods, 24 guide tubes and 1 instrumentation tube. Figure 2-2 below shows the cross sectional view of a North Anna fuel assembly.

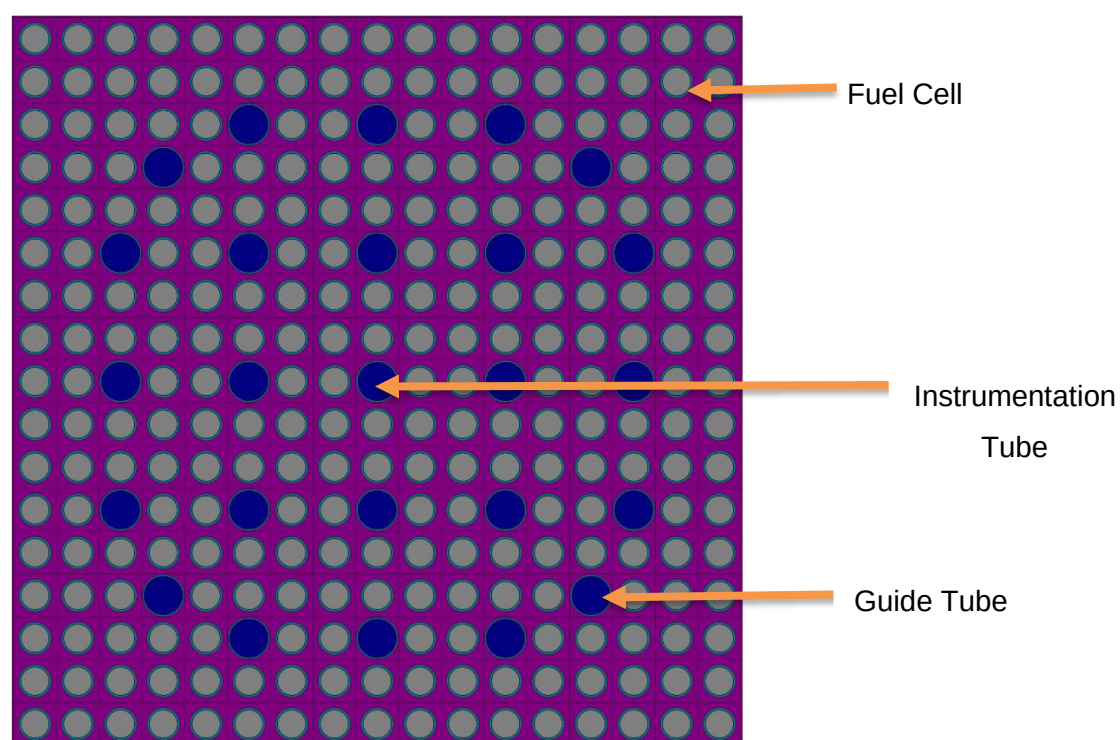


Figure 2-2: Cross Sectional View of a North Anna Fuel Assembly

The design parameters for the North Anna fuel assemblies are given in Table 2-4 below.

Table 2-4: Properties of North Anna Fuel Assemblies (VEPCO, 2009)

Properties	Values
No of Fuel Rods	264
No of Guide Tubes	24
No of Instrumentation Tubes	1
Rod Pitch	1.25894 <i>cm</i>

2.3.5 Fuel Cells

The North Anna fuel rods are made up of a Zircaloy cladding which encloses the fuel pellets. There is a helium gap between the cladding and the fuel pellets. A cross sectional view of a North Anna fuel cell is shown in Figure 2-3 below.

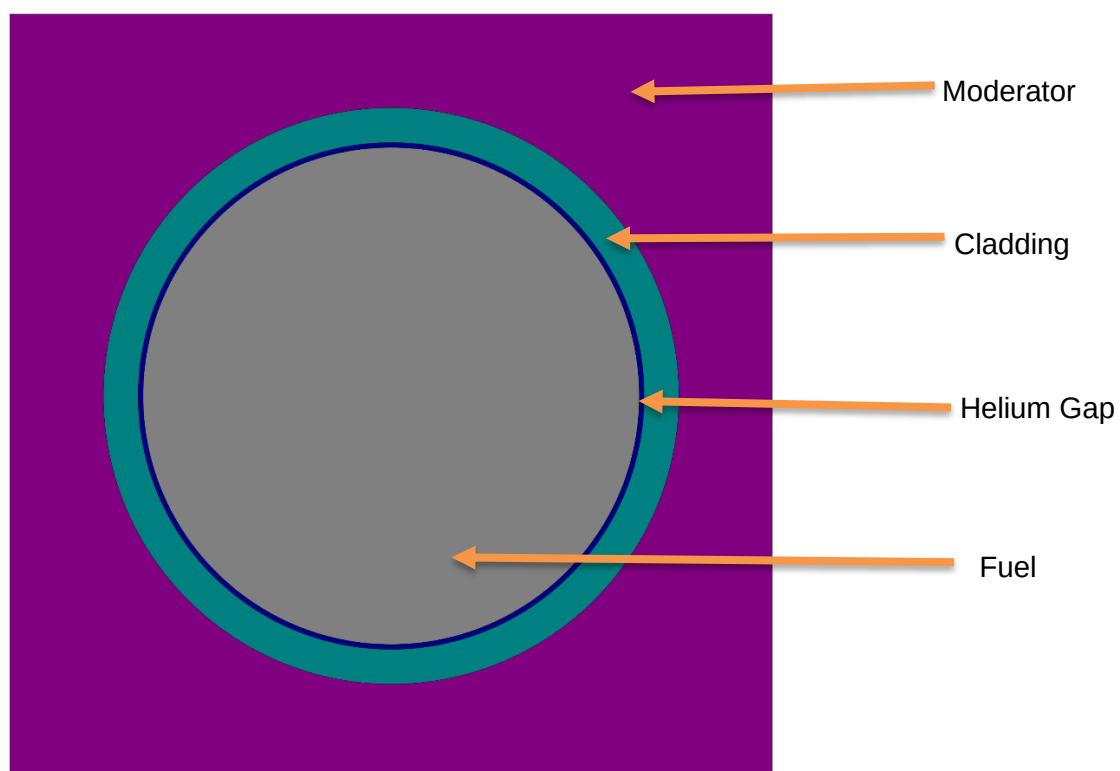


Figure 2-3: Cross Sectional View of a North Anna Fuel Cell

Table 2-5 below lists the material properties and dimensions of the North Anna fuel cells. The average temperatures for the different materials in the fuel cells are given in Table 2-6.

Table 2-5: Properties of the North Anna Fuel Cells (VEPCO, 2009) (ATI, 2019)

Properties	Values
Fuel Cell Width	1.25894 <i>cm</i>
Fuel Cell Breadth	1.25894 <i>cm</i>
Fuel (UO_2) Density	10.970 <i>g/cm³</i>
Fuel (UO_2) Enrichments	2.1 w/o, 2.6 w/o and 3.1 w/o
Fuel Pellet Outer Radius	0.40958 <i>cm</i>
Helium Density	0.0016038 <i>g/cm³</i>
Cladding (Zircaloy-4) Density	6.5500 <i>g/cm³</i>
Cladding Inner Radius	0.41783 <i>cm</i>
Cladding Outer Radius	0.47498 <i>cm</i>
Moderator (H_2O) Density	0.64990 <i>g/cm³</i>

Table 2-6: Temperatures of the Fuel Cell Materials (VEPCO, 2009)

Materials	Temperatures (K)
Fuel (UO_2)	1200
Helium Gap	900
Cladding (Zircaloy-4)	700
Moderator (H_2O)	500

2.3.6 Guide Tubes and Control Rods

The 24 guide tubes in the North Anna assemblies serve as insertion points for control rods. If control rods need to be inserted into the fuel assemblies, they can be lowered into the guide tubes from above the core. The guide tubes are made from Zircaloy-4. The control rods consist of a 304 Stainless Steel cladding, which encloses the neutron poison (80 w/o Ag, 15 w/o In, 5 w/o Cd). Figure 2-4 below shows a cross sectional view of the North Anna guide tube with a control rod inserted.

The material properties of the guide tubes and control rods are listed in Table 2-7, while their temperatures are listed in Table 2-8.

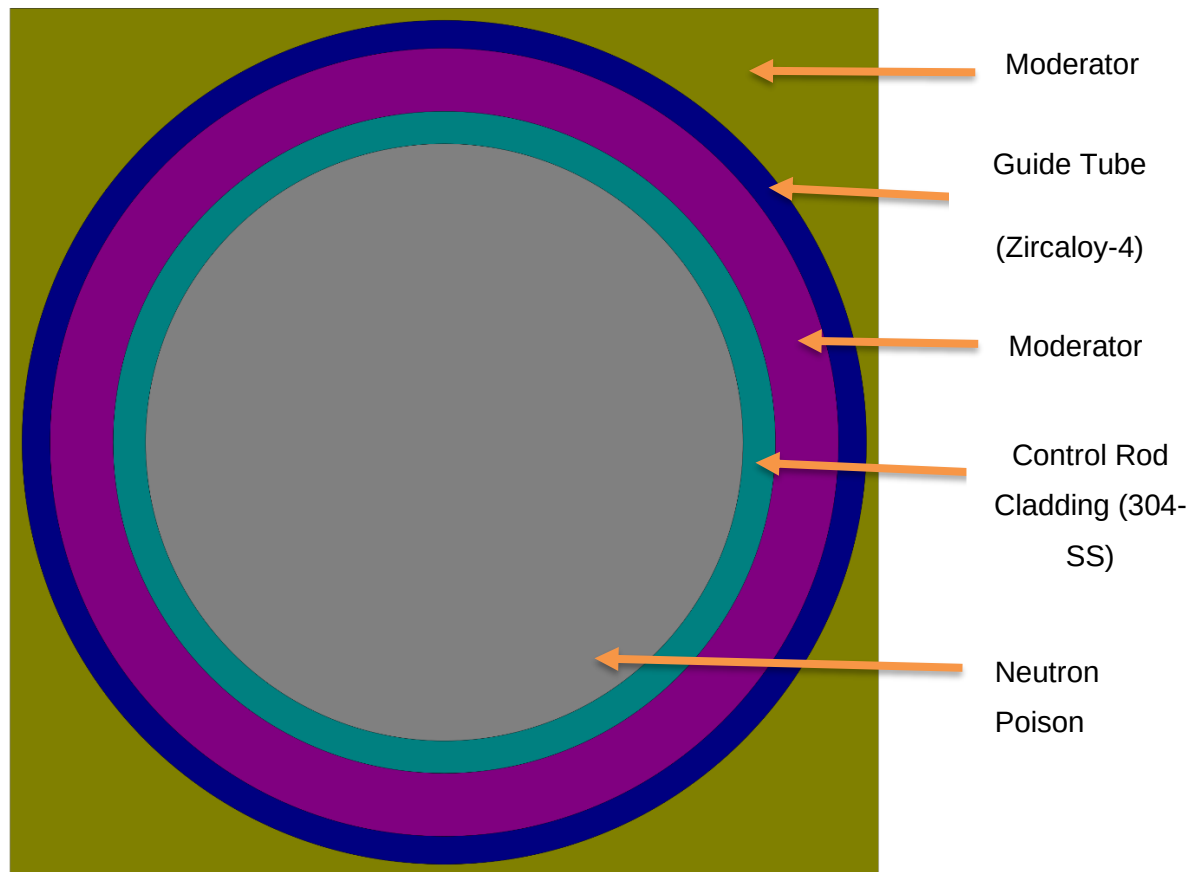


Figure 2-4: Cross Sectional View of Guide Tube with Control Rod Inserted

Table 2-7: Material Properties of the Guide Tubes and Control Rods (VEPCO, 2009)

Properties	Values
Neutron Poison Density (80w/o Ag, 15w/o In, 5w/o Cd)	10.160 g/cm^3
Neutron Poison Outer Radius	0.43307 cm
Control Rod Cladding (304-SS) Density	8.0000 g/cm^3
Control Rod Outer Radius	0.48006 cm
Guide Tube Cladding (Zircaloy-4) Density	6.5500 g/cm^3
Guide Tube Inner Radius	0.57150 cm
Guide Tube Outer Radius	0.61214 cm

Table 2-8: Guide Tube and Control Rod Temperatures (VEPCO, 2009)

Material	Temperatures (K)
Neutron Poison (80 w/o Ag, 15 w/o In, 5 w/o Cd)	550
Control Rod Cladding (304-SS)	550
Moderator	550
Guide Tube Cladding	550

CHAPTER 3 METHODOLOGY

3.1 Compiling and linking the DRAGON and DONJON Source Codes

The source codes for both DRAGON and DONJON are readily accessible for download at the Polytechnique de Montreal website (Poltechnique, 2019) since they are open source. For both DRAGON and DONJON, the source code were downloaded in three separate folders: utility, GAN and main source files.

The Linux operating system was used. A compilation bash script was written in Notepad++ for each one of the three source code folders. Notepad++ is a text editor which uses a Windows operating system. The scripts were written such that they produced the object files in a common folder. Since the source files were written in Fortran, a Fortran compiler was called in the bash scripts (gfortran). The compilation scripts were then executed. It was noted that the GAN library folder contained two source files written in C. These were compiled separately using the relevant compiler (gcc).

The main DRAGON and DONJON source code files are DRAGON.f and DONJON.f. Linking bash scripts were written in Notepad++ to link the other object files linked with these main source code files in order to create working executable programs for DRAGON and DONJON.

The DRAGON and DONJON executables were tested on reference DRAGON and DONJON models which are provided by the DRAGON and DONJON development teams respectively (Poltechnique, 2019). The multiplication constants of the reference models were calculated in both DRAGON and DONJON. For verification, the multiplication constants calculated with the executables were compared to the multiplication constants supplied for the reference models by the development teams.

3.2 North Anna Fuel Assembly Models: DRAGON

The North Anna reactor core consists of three types of fuel assemblies which differ only by fuel enrichments as described in section 2.3. The DRAGON models of these assemblies were developed by making a series of modifications to a given DRAGON reference fuel cell model, which is described in Section 3.2.3.

The DRAGON models were verified in two ways. The first verification was visual, where the diagram for each model was generated in DRAGON and compared to the expected diagram

according to the North Anna specifications document (VEPCO, 2009). The second verification was numerical, where for each model, the infinite multiplication constant was compared to the infinite multiplication constants of the preceding model and the subsequent model. The changes in the multiplication constant between subsequent models were analyzed to determine whether those changes are accurately predicted by reactor physics theory.

Before describing the benchmark cell model and its modifications, an outline of the DRAGON input file is given next regarding syntax and structure.

3.2.1 DRAGON Input File Syntax

A DRAGON input file is a record of instructions for the execution of DRAGON. The file can be written using standard text editors such as Notepad++ and saved as a generic file format or text file. The instructions must be written within the first 72 columns of each line of the file. Entries beyond the 72nd column are automatically treated as comments. Comments may be entered in the first 72 columns by beginning a line with an asterisk (*) or an exclamation mark (!) (Marleau, et al., 2014).

As mentioned in section 2.1, DRAGON is a modular code which transfers information between the modules using data structures. All variables, data structures and modules used in a DRAGON input must first be declared by type. This is done by writing the name of the variable, data structure or module after its type keyword. For instance, all modules are declared by writing them after their keyword "MODULE". Similarly, each data-structure is declared by writing it after its type keyword. The data-structure types and thus keywords are summarized in Table 2-1. Each module is terminated with a semicolon ";".

A DRAGON execution takes the input file together with a cross-section library and produces various outputs which depend on the specifications made in the input file. One of the outputs is a record of the execution, which documents the input file in its entirety, together with all the steps that DRAGON has taken during the execution. This record is produced even when the execution was unsuccessful, listing the errors encountered.

3.2.2 General Structure of a DRAGON Input File

A DRAGON input file generally contains information about the geometry and material properties of the system being modelled, along with specifications about the type of calculations which are

to be carried out. The calculations as noted in section 2.1 include geometry tracking, self-shielding, transport calculations, group collapse and homogenization, burnup, model diagram production, as well as an extraction of the homogenized few group constants. Therefore, a general DRAGON input file has the form:

- I. Declaration of modules, data-structures and variables;
- II. Materials specifications;
- III. Geometry specifications;
- IV. Geometry tracking and drawing;
- V. Self-shielding;
- VI. Transport calculations;
- VII. Group collapse and homogenization;
- VIII. Extraction of the few-group, homogenized cross-sections;
- IX. Ending.

I. Declaration of Modules, Data-structures and Variables

All modules, data structures and variables must be declared before being used in the input file, as outlined in section 3.2.1. For instance, a DRAGON input file which features the geometry module will also feature the data structure associated with the module as well as any variables used. The declaration of the module and its data structure would take the form:

```
MODULE GEO:
LINKED_LIST NORTHANNA
INTEGER i j k
```

In this example, the data structure is named “NORTHANNA” and declared as a “LINKED_LIST”. The variables i, j and k are declared as integers.

II. Materials Specifications

The material properties of a system being modelled in DRAGON can be handled by two modules, MAC: and LIB:. The module MAC: accepts macroscopic cross-sections entered directly by the user during a DRAGON execution. The module LIB: on the other hand, retrieves microscopic

cross-sections from a cross-section library (such as ENDF) and requires the user to specify the material number densities and temperatures.

The module LIB: has the advantage of simplifying the materials specification process. MAC: requires the user to first acquire microscopic cross-sections for each isotope (and each type of reaction), perform temperature interpolations, calculate the number densities and finally the macroscopic cross-sections. LIB:, in contrast, only requires the names of the isotopes in the cross-section library, their temperatures and number densities. It was decided to use the LIB: format in this work, with the implementation of MAC: reserved for a future study. This discussion is thus focused on the module LIB:. A general materials specification with LIB: has the format:

```
LIBNAME := LIB: :: LIBOPTIONS ;
```

“LIBNAME” is the user-specified name of the LINKED_LIST data-structure which is generated by LIB: to store the material properties.

“LIBOPTIONS” include the name and type of the microscopic cross-section library, the composition, temperatures and number densities of mixtures, and their indices. The module is terminated with the semicolon “;”.

Example:

```
NACSXNS := LIB: ::  
  EDIT 2  
  NMIX 2 MIXS LIB: WIMSD4 FIL: iaea  
  MIX 1 1200.0  
    OXY16 = '6016' 0.004  
    UR235 = '2235' 0.00002  
  MIX 2 600.0  
    OXY16 = '6016' 0.0001  
    HYD1 = '3001' 0.0002  ;
```

In this example, the LINKED_LIST data-structure which stores the material properties is named “NACSXNS”. The keyword “EDIT” is used to set the print level on “NACSXNS”; 0 means no information will be printed onto “NACSXNS” while higher print levels increase information that is printed on the LINKED_LIST (this applies for all the modules).

There are two mixtures in the system as specified by “NMIX 2”. The microscopic cross-sections for their isotopes will be read from the library “iaea” which is in the WIMSD4 format, as denoted by “LIB: WIMSD4 FIL: iaea”. The two mixtures are indexed as 1 and 2 using the keyword “MIX”. Mixture 1 is at a temperature 1200.0 K and is composed of two isotopes which the user has named “OXY16” and “UR235”. Their indices in the library “iaea” are 6016 (O16) and 2235 (U235) respectively. Their number densities (atoms/cc-barn) are 0.004 and 0.00002 respectively. Mixture two follows the same format.

III. Geometry Definition

The geometry definition of a model includes the type of geometry, dimensions and where different mixtures are in the geometry. The module GEO: is responsible for handling these specifications. A geometry specification takes the format:

```
GEONAME := GEO: :: GEOOPTIONS ;
```

“GEONAME” is a name chosen by the user for the LINKED_LIST which is created by GEO: to store the geometry specifications.

“GEOOPTIONS” include the type of geometry, dimensions, boundary conditions and location of mixtures in the geometry.

Example:

```
NAGEO := GEO: :: CARCEL 1
EDIT 2
X- REFL  X+ REFL
Y- REFL  Y+ REFL
MESHX 0.0 3.0
MESHY 0.0 4.0
RADIUS 0.0 1.0
MIX 1 2 ;
```

In this example the LINKED_LIST is named “NAGEO”. The geometry is specified as a 2-dimensional Cartesian cell with an embedded cylinder by the keyword “CARCEL 1”. Reflective

boundary conditions are specified for the four sides of the Cartesian cell: X- (left), X+ (right), Y- (bottom) and Y+ (top) using the keyword “REFL”.

The keywords “MESHX” and “MESHY” specify the dimensions of the geometry along the x and y axes. In this example the Cartesian cell extends from 0.0 to 3.0 cm in the x-axis and from 0.0 to 4.0 cm along the y-axis. The keyword “RADIUS” specifies the radial dimensions, in this case the radius of the embedded cylinder is set to 1 cm. The last line of the example “MIX 1 2” denotes that mixture 1 occurs within the embedded cylinder, while mixture 2 occurs in the cell outside the cylinder.

The DRAGON models in this research project were specified in 2D since the assemblies are assumed to have infinite height. A 2D plane through an infinitely high assembly is sufficient to investigate the neutronic properties of the assemblies, since they remain constant along the assembly’s height.

IV. Geometry Tracking and Drawing

As outlined in section 2.1, the five DRAGON tracking modules process the geometry defined in GEO:. A general tracking specification has the form:

```
TRACKNAME TRACKFILE := TRACKMODULE: GEONAME ::  
  
TRACKINGOPTIONS ;
```

“TRACKNAME” is a user-specified name of the LINKED_LIST which is created by the tracking module to store tracking information.

“TRACKFILE” is also named by the user. It is an XSM_FILE which stores tracking lengths. This data structure is only compulsory for the tracking module EXCELT:.

“TRACKMODULE:” stands for one of the five tracking modules listed in Table 2-1.

“TRACKINGOPTIONS” depend on the tracking module selected. Generally, these include the option of setting the print level on the output data structures and setting the name of the directory in “TRACKNAME” in which the tracking information is stored.

Example:

```
NATRACK NAFILE := EXCELT: NAGEO ::  
    EDIT 2  
    TITLE 'NORTH ANNA FUEL CELL'  
    MAXR 3 ;
```

In this example the XSM_FILE is named “NAFILE” and the LINKED_LIST is named “NATRACK”. The EXCELT: tracking module is used for tracking the geometry “NAGEO”. The keyword “TITLE” is for setting the name of the directory on “NATRACK” in which the tracking information is stored, in this case the directory is named “NORTH ANNA FUEL CELL”. The last line of the example sets the maximum number of regions which can be expected in the geometry, in this case it is set to 3.

The tracking information stored in the LINKED_LIST “NATRACK” can be used by the graphics modules in DRAGON to produce a diagram of the geometry. There are two graphics modules in DRAGON, TLM: and PSP:. The module TLM: is only compatible with the tracking module NXT:. It takes as input NXT: tracking information and produces a MATLAB m-file (Marleau, et al., 2014) which contains the plotting instructions for the geometry diagram. The module PSP: on the other hand, is compatible with both NXT: and EXCELT:. It produces a POSTSCRIPT format graphical file of the geometry.

The PSP: module is discussed here since it accommodates both EXCELT: and NXT:. A general PSP: specification takes the following form:

```
DIAGRAM.ps := PSP: TRACKNAME :: GRAPHICSOPTIONS ;
```

The PSP: output data structure “DIAGRAM.ps” is a SEQ_ASCII file which contains the diagram of the geometry.

“GRAPHICSOPTIONS” include setting the colour scheme for the graphics files.

Example:

```
NAGRAPH.ps := PSP: NATRACK ::  
    FILL RGB
```

TYPE MIXTURE ;

In this PSP: example, the graphics file is named “NAGRAPH.ps”. The keyword “FILL” is for setting the colour scheme for the graphics file, in this case it is set as “RGB” which stands for Red-Green-Blue. There is also the option CMYK which stands for Cyan-Magenta-Yellow-Black. The last line “TYPE MIXTURE” specifies that regions which contain the same mixture are assigned the same colour.

V. Self-Shielding

Self-shielding is handled by the module SHI: in DRAGON. This module takes the LINKED_LIST files generated by LIB: and the tracking module as input. A general specification for this module has the form:

LIBNAME2 := SHI: LIBNAME TRACKNAME :: SHIELDINGOPTIONS ;

“LIBNAME2” is the self-shielded version of “LIBNAME”.

“SHIELDINGOPTIONS” include setting the print level of the module.

Example:

NALCSXNS2 := SHI: NACSXNS NATRACK:: EDIT 3 ;

“NACSXNS2” is the self-shielded version of “NACSXNS”.

In this research project, the cross-section library that was used for the DRAGON models is a 69-group IAEA cross-section library which is supplied by the DRAGON and DONJON development teams (Poltechnique, 2019). The IAEA cross-section library contains self-shielded cross-sections which were most likely prepared using typical light water fuel pins. Using the cross-section library for the fuel pins of the North Anna reactor is thus reasonable.

However, for other materials such as the neutron poisons, the self-shielding calculations must be done specifically for that material directly from a continuous energy spectrum evaluated nuclear

data file. It is therefore acknowledged that the 69-group IAEA cross-section library may not produce the best results for the materials of the DRAGON models.

VI. Transport Calculations

As discussed in section 2.1, the assembly modules ASM: and EXCELL: prepare the collision probability matrices which are required by the module FLU: for solving the transport equation. The calculation for the collision probability matrices generally takes the format:

```
CPNAME := ASSMEBLYMODULE: LIBNAME2 TRACKNAME TRACKNAMEFIL ::  
    ASSEMBLYOPTIONS ;
```

“CPNAME” is the LINKED_LIST data-structure which stores the calculated matrices.

“ASSEMBLYMODULE:” is either ASM: or EXCELL:

“ASSEMBLYOPTIONS” include setting the print level for “CPNAME”.

Example:

```
NACP := ASM: NACSXNS2 NATRACK NAFIL :: EDIT 2 ;
```

In the above example, the ASM: LINKED_LIST is named “NACP”. The LINKED_LIST NACP is then taken as input by FLU: in a transport calculation. Below is a general format of the transport calculation:

```
FLUNAME := FLU: CPNAME LIBNAME2 TRACKNAME :: FLUXOPTIONS ;
```

“FLUNAME” is the name of the FLU: data structure which stores the transport solutions.

“FLUXOPTIONS” includes the type of solution which is determined and setting the print level.

Example:

```
NAFLU := FLU: NACP NACSXNS2 NATRACK ::
```

```
EDIT 2
TYPE K ;
```

In this example the FLU: LINKED_LIST is named "NAFLU". The keyword "TYPE" is for specifying the type of calculation which is to be performed. In this case a search for the multiplication constant is done as denoted by the keyword "K". A search for the buckling constant would be denoted as "TYPE B".

VII. Group-Collapse and Homogenization

Group collapse and homogenization are handled by the module EDI: in DRAGON. This module admits as input the FLU:, LIB: and tracking data-structures. A general format for an EDI: calculation looks as follows:

```
HOMCSXN := EDI: FLUNAME LIBNAME2 TRACKNAME ::
EDIOPTIONS ;
```

"HOMCSXN" is the name of the EDI: LINKED_LIST which stores the homogenized few-group cross-sections.

"EDIOPTIONS" includes setting the print level for the module, setting the energy limits for the few groups and specifying the level of homogenization which is performed.

Example:

```
NAEDI := EDI: NAFLU NACSXNS2 NATRACK ::
EDIT 2
MERGE COMP
COND 0.625
SAVE ;
```

In this example, the EDI: LINKED_LIST is named "NAEDI". The keyword "MERGE" is for specifying the level of homogenization. The geometry can either be homogenized completely into a single effective material, or only partially. In the latter, some regions are merged with some but not with others, such that the final geometry consists of more than one mixture after

homogenization. In this example complete homogenization is performed, as denoted by the keyword “COMP”.

The keyword “COND” is for specifying the energy limits between the new energy groups. In this example the endpoint energy values (Emin and Emax) in NACSXNS2 are implied while the spectrum is condensed into two groups, group 2 (minimum energy to 0.625 eV) and group 1 (from 0.625 eV to maximum energy). The keyword “SAVE” is for saving the 2 group homogenized cross-sections on “NAEDI” in the directory ‘REFCASE 1’. This directory is created automatically by DRAGON.

VIII. Extracting the Few-group, Homogenized Cross-sections

CPO: is the dedicated DRAGON module for extracting the few-group homogenized cross-sections from the EDI: LINKED_LIST and writing them into an XSM_FILE. For the LINKED_LIST NAEDI created in the EDI: example in section VII, the extraction specification would take the following form:

```
COMPO.FILE := CPO: NAEDI ::  
    STEP 'REF-CASE 1'  
    EXTRACT ALL NAME COMPO ;
```

where “COMPO.FILE” is the XSM_FILE which stores the extracted cross-sections.

The keyword “STEP” is to specify the directory in “NAEDI” from which the cross-sections are extracted. In this case the directory is ‘REF-CASE 1’. The last line specifies that “ALL” the cross-sections in ‘REF-CASE 1’ are “EXTRACTED” and written on the XSM_FILE of “NAME COMPO”. The significance of this file is that it is a separate output of the overall DRAGON execution. It can be used as an input for DONJON.

IX. Ending the DRAGON Input File

The END: module is dedicated to ending a DRAGON execution. It must also be declared with the rest of the modules in the input file. It is called at the very end of the input file, and must be terminated with a semicolon as follows:

```
END: ;
```

3.2.3 The Reference Model: TCWU01 – Case 4

TCWU01 is a DRAGON input file supplied by the DRAGON development team (Poltechnique, 2019). It is provided along with its output so that users can compare their output for TCWU01 with the provided output to determine if their installation works as intended. A 69-energy-group WIMSD4 microscopic cross-section library is also provided for the input file. In this library, the self-shielding calculations were already completed, and were thus not required in the DRAGON calculation.

TCWU01 features two Mosteller (Mosteller, 1991) fuel cell models, an annular 1-dimensional cell and a Cartesian 2-dimensional cell. It features four calculation cases for the two geometries; cases 1 and 3 for the annular cell and cases 2 and 4 for the Cartesian cell. The choice of a tracking module is the difference between cases 1 and 3. The same is true for cases 2 and 4. Cases 1 and 2 feature the JPMT: tracking module while cases 3 and 4 feature the EXCELT: tracking module.

The 2-dimensional fuel cell was chosen as the reference model for the DRAGON models in this research project since it closely resembles the fuel cells of the North Anna reactor. Case 4 of the 2-dimensional fuel cell model was chosen for the advantages of the tracking module EXCELT: over JPMT: as outlined in section 3.3.2. The material properties and dimensions of TCWU01 - case 4 are summarized in Table 3-1 and Table 3-2 respectively. As noted in Table 3-1 the fuel is 0.711 wt.% enriched uranium dioxide. A diagram of the model is given in Figure 3-1.

Table 3-1: Material properties of TCWU01 – case 4 (Marleau, et al., 2014)

Material	Chemical Formula	Number density (10^{24} molecules/cm ³)	Temperature (K)
Fuel (0.711 wt.%)	UO_2	0.0230655	600
Cladding	Zr	0.0383243	600
Moderator	H_2O	0.0221163	600

Table 3-2: Dimensions of TCWU01 – case 4 (Marleau, et al., 2014)

Dimension	Value (cm)
Cell Length	1.26209
Cell Breadth	1.26209
Cladding Outer Radius	0.45802
Fuel Pellet Radius	0.39306

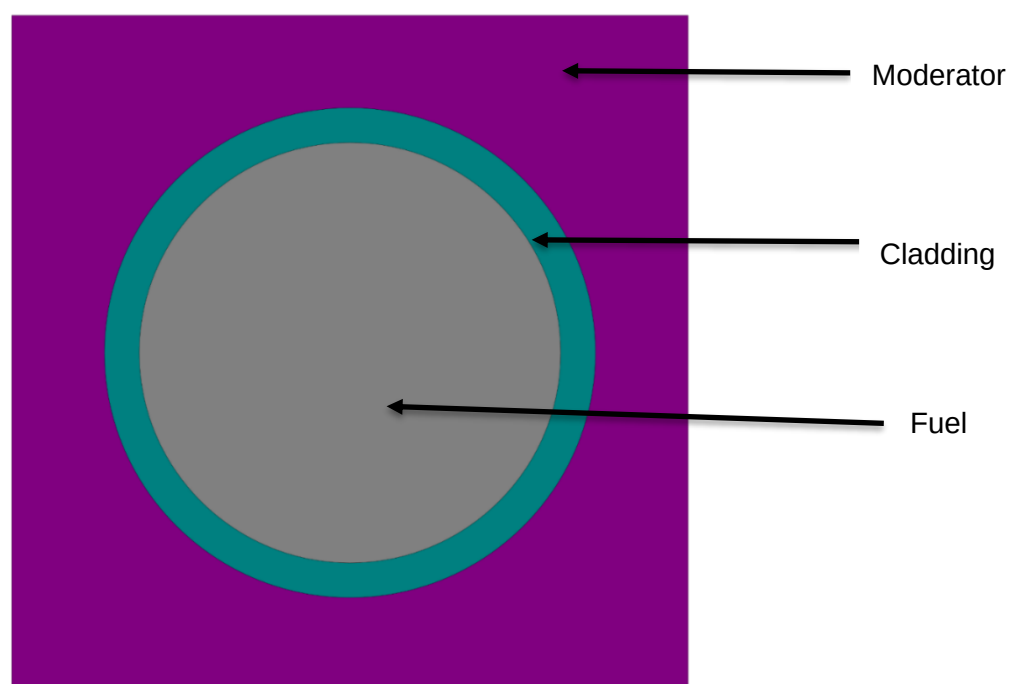


Figure 3-1: TCWU01 – case 4

3.2.4 From TCWU01 – Case 4 to the North Anna Assembly Models

The North Anna fuel assembly models were developed by making a series of modifications to TCWU01 – case 4. The modifications were divided into two stages. The first stage modifications

involved changing the test case fuel cell into a North Anna fuel cell. The second stage modifications involved developing the North Anna fuel assembly models from the North Anna fuel cell developed in the first stage. Table 3-3 summarizes the modifications from the reference model to the three North Anna fuel assembly models. Figures 3-2 to 3-7 show the geometry changes between subsequent modules.

Table 3-3: Development of the North Anna Assembly Models

Model Name	Preceding Model	Modification to Preceding Model	Diagram
Drag_0	TCWU01 – case 4	Boron was removed from the moderator.	Figure 3-2
Drag_1	Drag_0	Cell dimensions were changed to 1.25894 cm	Figure 3-3
Drag_2	Drag_1	Helium gap was added by changing the cladding inner and outer radii to 0.41783 cm and 0.47498 cm respectively.	Figure 3-4
Drag_3	Drag_2	Fuel pellet radius was changed to 0.40958 cm.	Figure 3-5
Drag_4	Drag_3	Fuel enrichment was changed to 2.1 w/o.	Figure 3-5
Drag_5	Drag_4	Moderator number density was changed to 0.021650×10^{24} molecules/cm ³ .	Figure 3-5
Drag_6	Drag_5	Cladding number density was changed from	Figure 3-5

		3.83243x10 ²² molecules/cm ³ to 0.0425x10 ²⁴ molecules/cm ³	
Drag_7	Drag_6	Fuel temperature was changed to 1200 K	Figure 3-5
Drag_8	Drag_7	Moderator temperature was changed to 550 K	Figure 3-5
Drag_9	Drag_8	Cladding and Helium gap temperatures were changed to 600 K and 900 K respectively	Figure 3-5
Drag_10	Drag_9	A 17 by 17 fuel assembly was constructed with the fuel cell Drag_9	Figure 3-6
Drag_11	Drag_10	An additional 0.0508 cm of moderator was added along the edges of the assembly to account for the moderator gaps between adjacent assemblies in the core.	Figure 3-7
Drag_12	Drag_11	25 fuel pins were replaced with guide thimble; the fuel pellet was removed while the cladding inner and outer radii were changed to 0.5715 cm and 0.61214 cm respectively. No control rods were inserted into the guide thimbles.	Figure 3-8
Drag_13	Drag_12	The energy limit between the two groups in the group collapse module was changed to 0.625 eV.	Figure 3-8

Drag_14	Drag_13	Fuel enrichment was changed to 2.6 w/o.	Figure 3-8
Drag_15	Drag_14	Fuel enrichment was changed to 3.1 w/o.	Figure 3-8

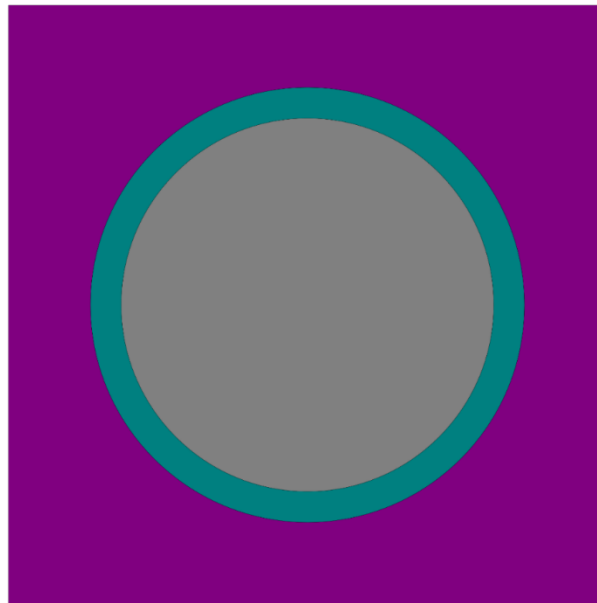


Figure 3-2: DRAG_0

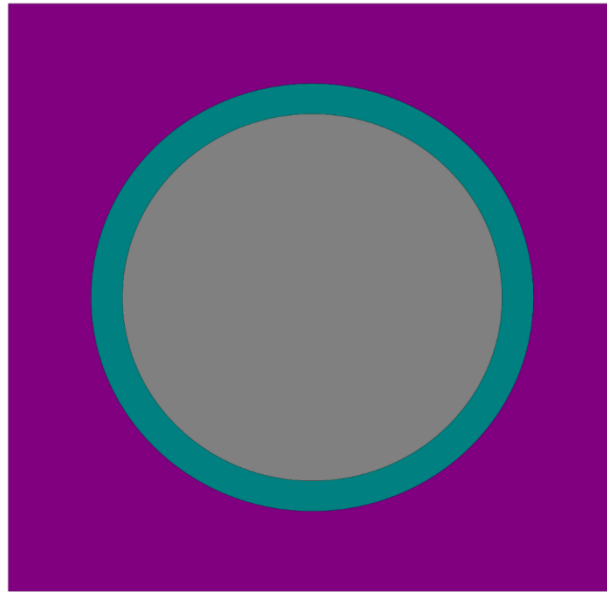


Figure 3-3: DRAG_1

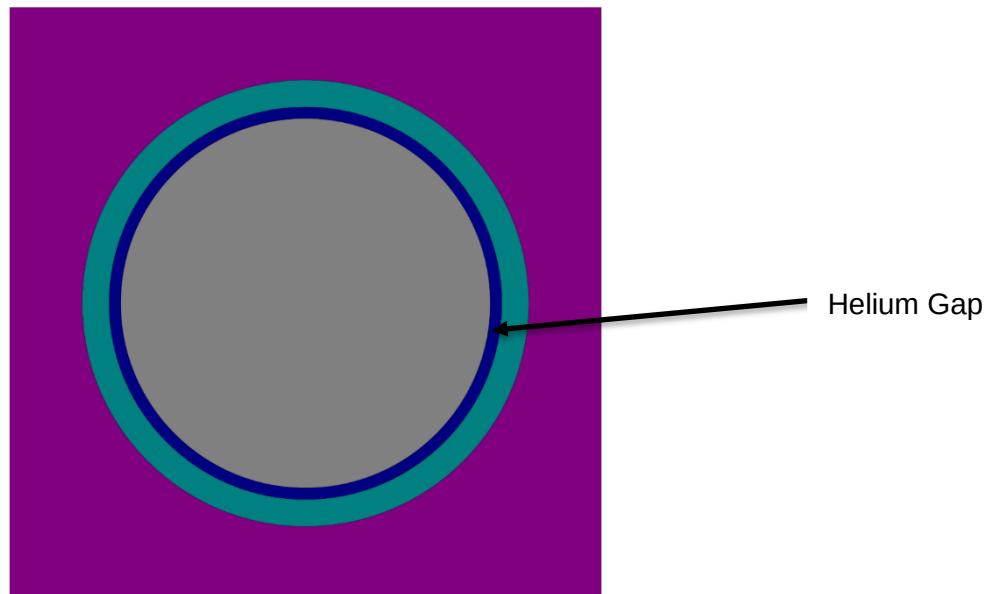


Figure 3-4: DRAG_2

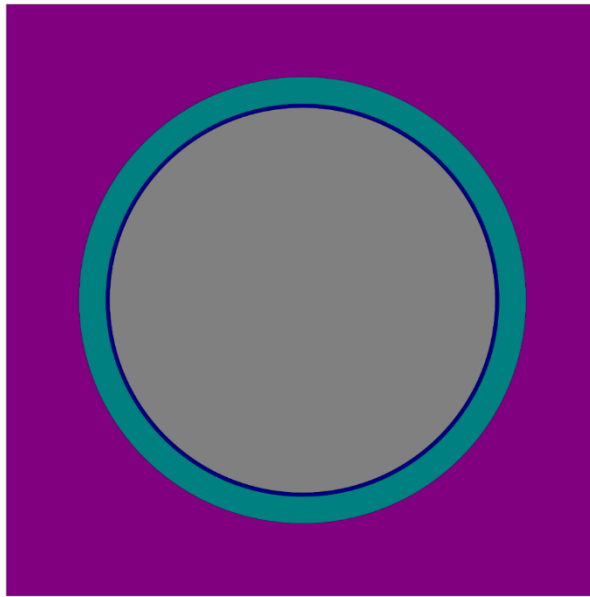


Figure 3-5: DRAG_3, DRAG_4, DRAG_5, DRAG_6, DRAG_7, DRAG_8, and DRAG_9

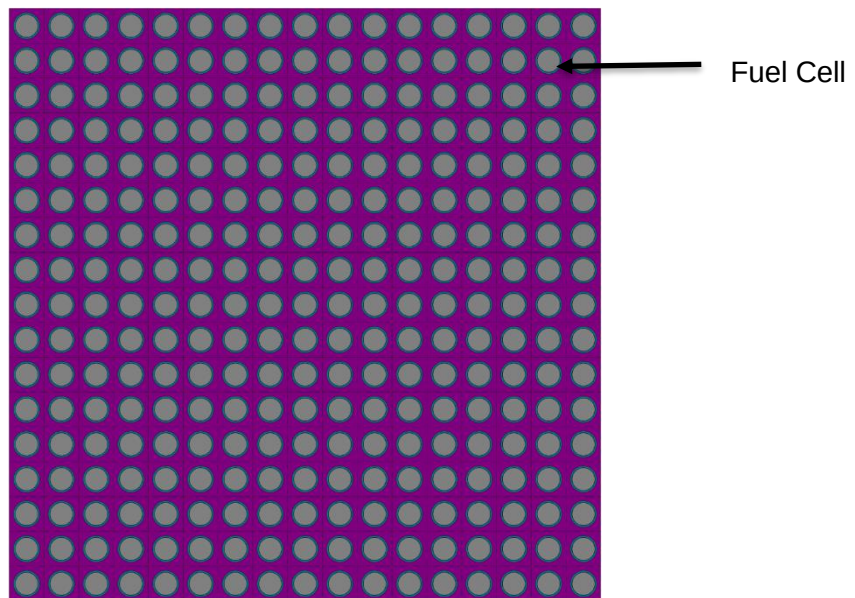


Figure 3-6: DRAG_10

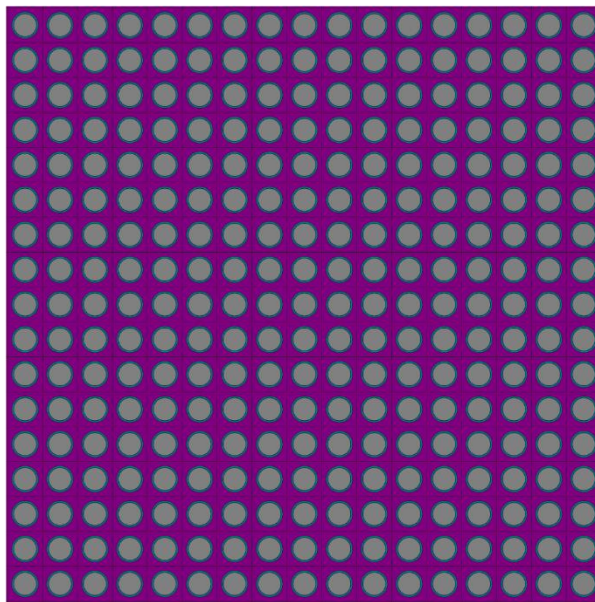


Figure 3-7: DRAG_11

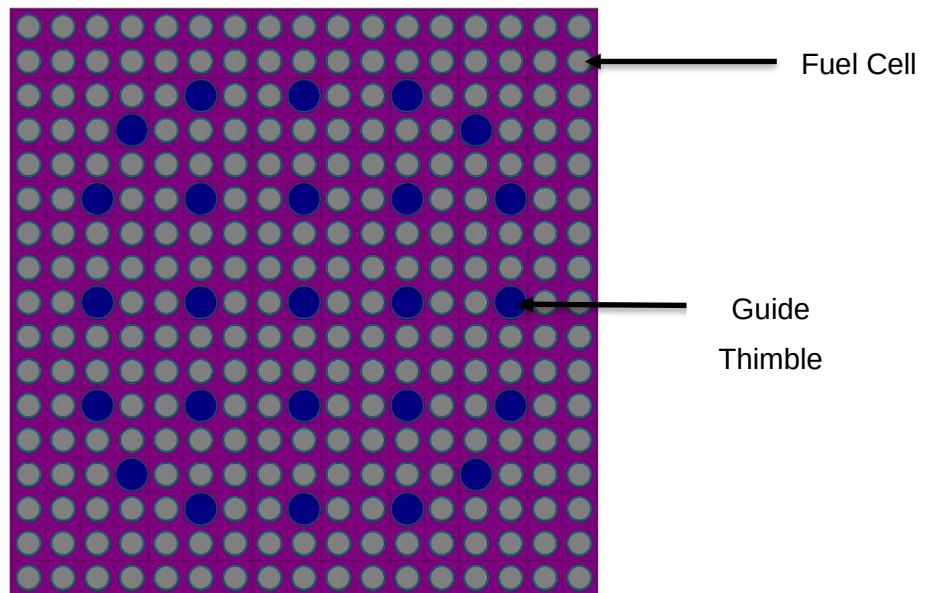


Figure 3-8: DRAG_12, DRAG_13, DRAG_14 and DRAG_15

3.3 The North Anna Full Core Model: DONJON

The North Anna reactor core model was developed based on a given DONJON reference model. The model was developed under the assumption that the core ends at the edge of each outer fuel assembly. Modelling the moderator around the core was outside the scope of this research project and will be considered as part of future work.

Void boundary conditions were specified for the DONJON models to account for the finite size of the reactor core. Although it was assumed that the core is surrounded by empty space, the void boundary condition still holds for the external surfaces of the core. In other words, the void boundary conditions specified for the empty assemblies at the corners of the core (see Figure 3-19) apply to the 3.1 w/o assemblies next to the empty assemblies.

The North Anna reactor core consists of three types of fuel assemblies which differ only by fuel enrichments, as described in Section 2.3. The DONJON model of the North Anna core was developed by making a series of modifications to a given DONJON reference core model. The DONJON models were verified in two ways.

Firstly, the diagrams of the models were generated and compared to the expected diagrams to make sure that the right materials are in the right positions. Secondly, for each model, the multiplication constant compared to the multiplication constant of the preceding and the subsequent models. The changes in the multiplication constant from one model to the next were analyzed to determine whether they were expected according to known reactor physics principles. Further, as was with the case for the DRAGON calculations, it also allowed a verification test of the building of the model using the input manual.

Before describing the benchmark cell model and its modifications, an outline of the DONJON input file is given regarding syntax and general structure.

3.3.1 DONJON Input File Syntax

A DONJON input file is a record of instructions for the execution of DONJON. The file can be written using standard text editors such as notepad++ and saved as a generic file format or as a text file. The instructions must be written within the first 72 columns of each line of the file. Entries beyond the 72nd column are automatically treated as comments. Comments may be entered in the first 72 columns by beginning a line with an asterisk (*) or an exclamation mark (!) (Varin, et al., 2005).

As mentioned in Section 2.2, DONJON is a modular code which transfers information between the modules using data structures. All variables, data structures and modules used in a DONJON input must first be declared by type. This is done by writing the name of the variable, data structure or module after its type keyword. For instance, all modules are declared by writing them after their keyword "MODULE". Similarly, each data-structure is declared by writing it after its type keyword. Each module is terminated with a semicolon ";".

A DONJON execution takes the input file together with a cross-section library and produces various outputs which depend on the specifications made in the input file. One of the outputs is a record of the execution, which documents the input file in its entirety, together with all the calculations that DONJON took during the execution. This record is also produced even when the execution was unsuccessful, listing the errors encountered.

3.3.2 General Structure of a DONJON Input File

A DONJON input file generally contains information about the geometry and material properties of the system being modelled, along with specifications about the type of calculations which are to be carried out. The calculations, as noted in Section 2.2, include geometry tracking, transport calculations, and burnup. Therefore, a general DONJON input file has the form:

- I. Declaration of Modules, Data-structures and Variables;
- II. Materials Specifications;
- III. Geometry Specifications;
- IV. Geometry Tracking ;
- V. Transport Calculations; and
- VI. Ending.

I. Declaration of Modules, Data-structures and Variables

All modules, data structures and variables must be declared before being used in the input file as outlined in Section 3.2.1. For instance, a DONJON input file which features the geometry module GEOD: will also feature the data structure associated with the module as well as any variables used. The declaration of the module and its data structure would take the form:

```
MODULE GEOD:
LINKED_LIST NAGEO
REAL o p q
```

In this example, the data structure is named “NAGEO” and declared as a “LINKED_LIST”. The variables o, p and q are declared as real numbers.

II. Materials Specifications

The material properties of a system being modelled in DONJON can be handled by two modules; MACD: and CRE:. The module MACD: accepts the material properties (macroscopic cross-sections, diffusion constants, temperatures etc.) entered directly by the user in the DONJON input file. The module CRE: on the other hand, retrieves material properties from one or more material properties files such as the one which is created by the DRAGON module CPO: as was shown in Section 3.2.2. The module CRE: has the advantage of simplifying the materials specification process. The DONJON material specifications for this research project were done in CRE:. A general materials specification with CRE: has the format:

```
LIBNAME := CRE: :: LIBOPTIONS ;
```

“LIBNAME” is the user-specified name of the LINKED_LIST data-structure which is generated by CRE: to store the material properties.

“LIBOPTIONS” include the names of material properties files, as well as the indexing of mixtures in the core. The module is terminated with a semicolon (;).

Example:

```
NALIB := CRE: LIBRARY1  LIBRARY2 ::  
  EDIT 2  
  NMIX 2  
  READ  COMPO    LIBRARY1  
          MIX 1    'DIRECTORY 1'    ENDMIX  
          COMPO    LIBRARY2  
          MIX 2    'DIRECTORY 1'    ENDMIX ;
```

In this example, the LINKED_LIST data-structure which stores the material properties is named “NALIB”. The keyword “EDIT” is used to set the print level on “NALIB”; 0 means no information

is printed onto "NALIB", while higher print levels mean more information is printed on the LINKED_LIST (this applies for all the modules).

There are two mixtures in the system as specified by "NMIX 2". They are indexed as 1 and 2 using the keyword "MIX". The material properties for these mixtures are read from the material properties files "LIBRARY1" and "LIBRARY2". Mixture 1 properties are read from "DIRECTORY 1" of "LIBRARY1", while properties of mixture 2 are read from "DIRECTORY 1" of "LIBRARY2". Each mixture definition is ended with the keyword "ENDMIX".

III. Geometry Definition

The geometry definition of a model includes the type of geometry, dimensions and where different mixtures are in the geometry. The module GEO: is responsible for handling these specifications. A geometry specification takes the format:

```
GEONAME := GEOD: :: GEOOPTIONS ;
```

"GEONAME" is a name chosen by the user for the LINKED_LIST which is created by GEOD: to store the geometry specifications.

"GEOOPTIONS" includes the type of geometry, dimensions, boundary conditions and location of mixtures in the geometry.

Example:

```
NAGEO := GEO:D :: CAR3D 4 1 2
EDIT 2
X- REFL    X+ REFL
Y- REFL    Y+ REFL
Z- REFL    Z+ REFL
MESHX 0.0  1.0  2.0  3.0  6.0
MESHY 0.0  4.0
MESHZ 0.0  20.0 30.0
MIX  PLANE  1
          1  2  1  2
```

```

    PLANE    2
          2  1  1  1
          ;

```

In this example the LINKED_LIST is named "NAGEO". A 3-dimensional Cartesian geometry is defined, as denoted by the keyword "CAR3D 4 1 2". The numbers "4 1 2" signify that the geometry is subdivided into 4 regions along the x-axis, 1 region along the y-axis and 2 regions along the z-axis. The keywords "MESHX", "MESHY" and "MESHZ" specify the mesh of the geometry along the x, y and z axes. For instance, the geometry is divided into four regions along the x-axis; the first region extends from 0.0 cm to 1.0 cm, the second region extends from 1.0 cm to 2.0 cm, the third region extends from 2.0 cm to 3.0 cm, and the fourth region extends from 3.0 cm to 6.0 cm.

Each subdivision along the z-axis is referred to as a plane. The planes are numbered from the top to the bottom of the z-axis. In the example above the core is divided into two regions in the z-axis. The top region is labelled as "PLANE 1" and the bottom plane is labelled as "PLANE 2". The thickness of each region is given by the z-axis mesh; the line "MESHZ 0.0 20.0 30.0" shows that the top plane extends from 0.0 cm to 20.0 cm while the bottom plane extends from 20.0 cm to 30.0 cm.

Mixtures are allocated to each region in each plane. In the top plane, mixture 1 is placed in the first and third regions while mixture 2 is placed in regions two and four, as denoted by the mixture sequence "1 2 1 2".

Reflective boundary conditions are specified for the six sides of the Cartesian geometry using the keyword "REFL", where for instance X- and X+ represent the left and right surfaces of the cartesian geometry.

IV. Geometry Tracking

As outlined in Section 2.2, the two DONJON tracking modules process the geometry defined in GEOD:. A general tracking specification has the form:

```

TRACKNAME := TRACKMODULE: GEONAME ::

TRACKINGOPTIONS ;

```

“TRACKNAME” is a user-specified name of the LINKED_LIST which is created by the tracking module to store tracking information for the geometry “GEONAME”.

“TRACKMODULE:” stands for one of the two DONJON tracking modules.

“TRACKINGOPTIONS” depend on the tracking module selected. Generally, these include the option of setting the print level on the output data structures and setting the name of the directory in “TRACKNAME” in which the tracking information is stored.

Example:

```
NATRACK NAFILE := TRIVAT: NAGEO ::  
    EDIT 2  
    TITLE 'NORTH ANNA CORE'  
    MAXR 10 ;
```

The LINKED_LIST in this example is named “NATRACK”. The TRIVAT: tracking module is selected for tracking the geometry “NAGEO”. The keyword “TITLE” is for setting the name of the directory on “NATRACK” in which the tracking information is stored, in this case the directory is named “NORTH ANNA CORE”. The last line of the example sets the maximum number of regions which can be expected in the geometry, in this case it is set to 10.

V. Multigroup Diffusion Calculations

As discussed in Section 2.2, the assembly modules BIVACA: and TRIVAA: prepare the collision probability matrices which are required by the flux solution module for solving the multi-group diffusion equation. The calculation for the collision probability matrices generally takes the format:

```
CPNAME := ASSMMOD: LIBNAME TRACKNAME ::  
    ASSMOPTIONS ;
```

“CPNAME” is the LINKED_LIST data-structure which stores the calculated matrices.

“ASSMMOD:” is either TRIVAA: or BIVACA:

“ASSMOPTIONS” include setting the print level for “CPNAME”.

Example:

```
NACP := TRIVAA: NALIB NATRACK :: EDIT 2 ;
```

In the above example, the TRIVAA: module is used. The TRIVAA: LINKED_LIST is named “NACP”.

The LINKED_LIST NACP is then taken as input by FLUD: in a transport calculation. Below is a general format of the transport calculation:

```
FLUNAME := FLUD: CPNAME LIBNAME TRACKNAME :: FLUXOPTIONS ;
```

“FLUNAME” is the name of the FLUD: data structure which stores the transport solutions.

“FLUXOPTIONS” includes specifying the type of solution which is determined and setting the print level.

Example:

```
NAFLU := FLUD: NACP NALIB NATRACK ::  
EDIT 2 ;
```

In this example the FLUD: LINKED_LIST is named “NAFLU”. The print level is set to 2.

VI. Ending the DONJON input file

The END: module is dedicated to ending a DONJON execution. It must also be declared with the rest of the modules in the input file. It is called at the end of the input file, and must be terminated with a semicolon as follows:

```
END: ;
```

3.3.3 The Reference Model: LMW43D

LMW43D is a DONJON input file supplied by the DONJON development team (Poltechnique, 2019). It is supplied together with its output file for users to test their DONJON installation; the user can run their DONJON installation with LMW43D and compare their output with the provided output.

The input file LMW43D models a 3-dimensional Cartesian reactor core. In the radial plane, $\frac{1}{4}$ of the core is modelled, and in the axial plane, $\frac{1}{2}$ of the core is modelled. The length and breadth of the core region modelled are divided into 6 subregions while the height is divided into 20 subregions. Table 3-4 summarizes the core dimensions and meshes. As outlined in Section 3.3.2, each z-axis subregion corresponds to a cross section through the core in the x-y plane. Figure 3-9 illustrates the top plane (plane 1) of the core. Planes 2 to 19 are identical. They are illustrated by Figure 3-10. The 20th plane is illustrated by Figure 3-11.

Void boundary conditions are applied to the x+, y+, z- and z+ edges of the core (In Figure 3-9, these are the right, bottom and top boundaries for the radial plane). Reflective boundary conditions are applied to the x- and y- edges of the core (In Figure 3-9, these are the left and right boundaries of the radial plane).

Geometry specifications in LMW43D are handled by GEOD:. The material properties are handled by MACD: as 2-group constants (macroscopic cross-sections, diffusion coefficients, etc.) entered directly in the input file for 5 mixtures. Only the 2-group constants are given for each mixture; no information about the type of fuel, cladding, moderator or their temperatures is given. Tracking and system matrix equation calculations are performed by TRIVAT: and TRIVAA: respectively. The multigroup diffusion calculations are performed by FLUD:.

Table 3-4: LMW43D Dimensions and Mesh (Varin, et al., 2005)

Dimension	Value(cm)	Mesh Array
Length(x)	110	0, 10, 30, 50, 70, 90, 110.
Breadth(y)	110	0, 10, 30, 50, 70, 90, 110.
Height(z)	200	0, 20, 27.5, 35, 42.5, 48.75, 55, 62.5, 70, 77.5, 85, 92.5, 100, 102.5, 110, 117.5, 125, 132.5, 140, 180, 200.

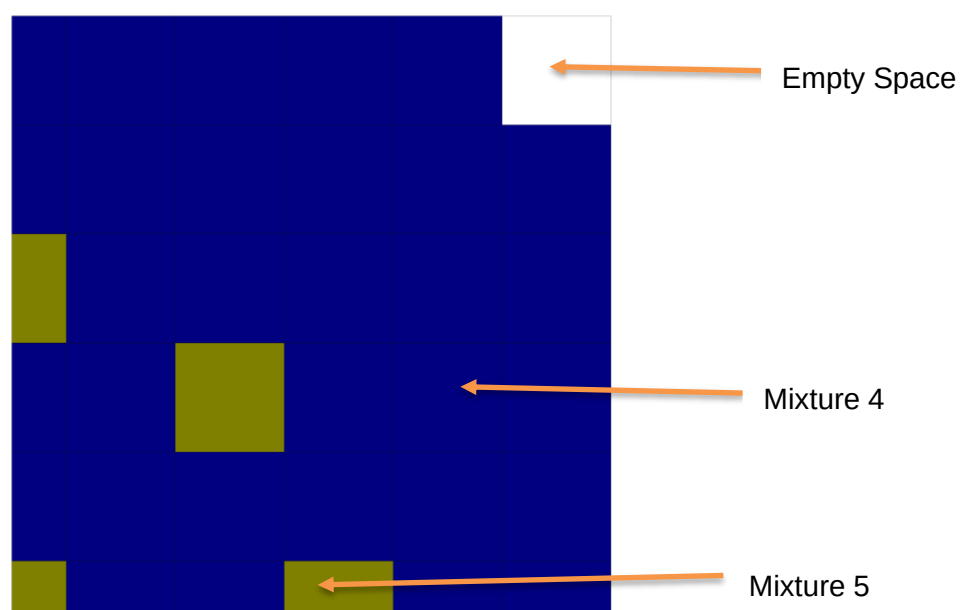


Figure 3-9: LMW43D Plane 1

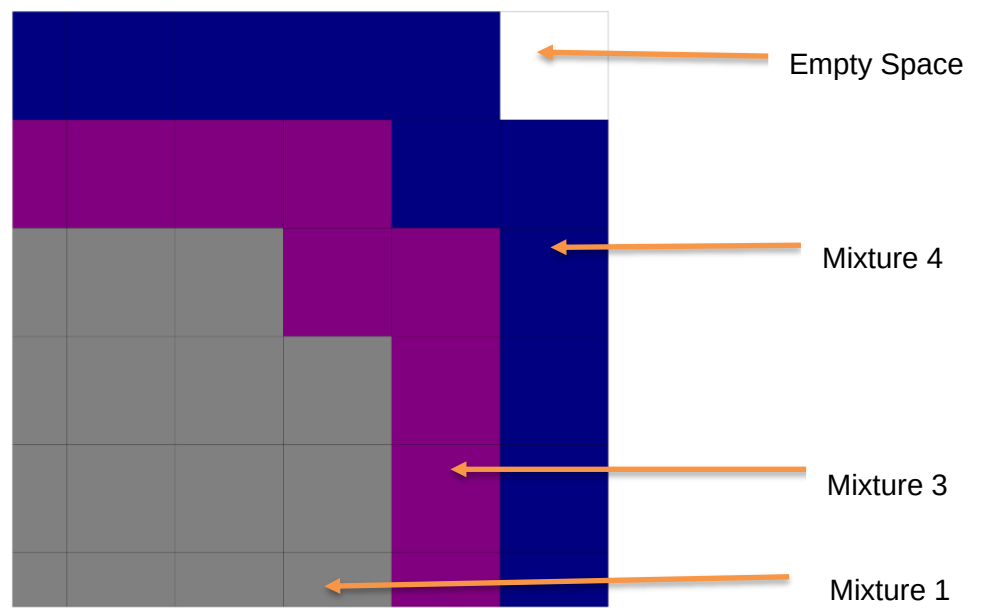


Figure 3-10: LMW43D Planes 2 to 19

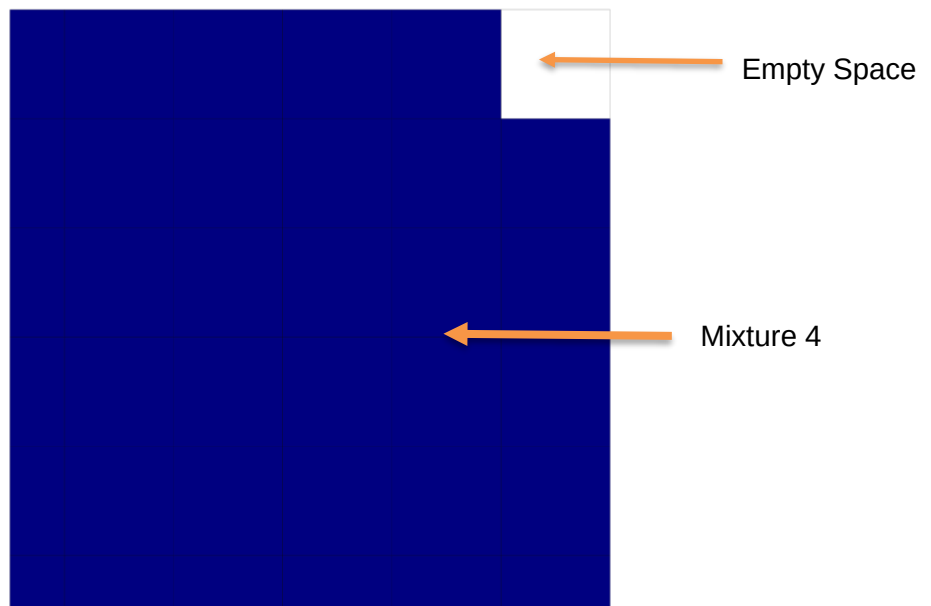


Figure 3-11: LMW43D Plane 20

3.3.4 From the Reference Model to the Full Core Model

The North Anna full core model was developed by making a series of modifications to the reference model LMW43D. Each modification yielded a new version of LMW43D. Each of the versions subsequent to LM43D were named “Don_i”, where i indexes the model version. For instance, Don_4 would be the fourth modified model. A summary of the modifications made to LMW43D in developing the full core model is given in Table 3-5. Graphical illustrations of the different stages of the model development process are also noted in Table 3-5.

The first modification to LMW43D was to replace the module MACD: with the module CRE: and replace the reflective boundary conditions with void boundary conditions. The reason for this modification was to enable DONJON to directly access the DRAGON output files for each of the three fuel assemblies.

Replacing the module MACD: also meant replacing the five mixtures which were specified under it. Under the module CRE:, material properties were read from the 3.1 wt% DRAGON output file such that the whole core consisted of one mixture. The resulting model was named Don_1.

The second, third and fourth modifications respectively were to change the x, y and z dimensions of the fuel assemblies in Don_1 to the dimensions of the North Anna fuel assemblies. The resulting models were named Don_2, Don_3 and Don_4 respectively.

The fifth, sixth and seventh modifications were to change the number of subdivisions along the x, y and z axes in Don_3 from 6, 6 and 20 respectively to 15, 15 and 15. These modifications were done to change the dimensions of the core in Don_3 to the dimensions of the North Anna reactor core. The resulting models were named Don_5, Don_6 and Don_7 respectively. It is noted that at this stage the $\frac{1}{4}$ core model was changed to a full core model. This is carried out to conform with the current methodology being adopted at the research unit. To have kept the $\frac{1}{4}$ core geometry would have been equally possibly.

The eighth modification was to remove the 3.1 wt% assemblies from the corners of the core and replace them with empty assemblies. This was done in accordance with the assumption that the core is surrounded by empty space. The resulting model was named Don_8.

The ninth and tenth modifications were to add 2.6 wt% and 2.1 wt% assemblies respectively into the core, replacing some of the 3.1 wt% assemblies. The assembly arrangement was done according to the North Anna Final Safety Analysis Report (VEPCO, 2009). The resulting models were named Don_9 and Don_10 respectively, such that Don_10 is the final North Anna core model.

Table 3-5: Development of the North Anna Full Core Model

Model Name	Preceding Model	Modification to Preceding Model	Diagram
Don_1	LMW43D	MACD: replaced with CRE:	Figure 3-12
Don_2	Don_1	Assembly x-dimensions changed to 21.50358 cm	Figure 3-13
Don_3	Don_2	Assembly y-dimensions changed to 21.50358 cm	Figure 3-14
Don_4	Don_3	Assembly z-dimensions changed to 24.3332 cm	Figure 3-14
Don_5	Don_4	Number of subdivisions along x-axis changed to 15	Figure 3-15
Don_6	Don_5	Number of subdivisions along y-axis changed to 15	Figure 3-16
Don_7	Don_6	Number of subdivisions along z-axis changed to 15	Figure 3-16
Don_8	Don_7	Corners of the core replaced with empty space	Figure 3-17
Don_9	Don_8	2.6wt% assemblies introduced into the core	Figure 3-18
Don_10	Don_9	2.1wt% assemblies introduced into the core	Figure 3-19

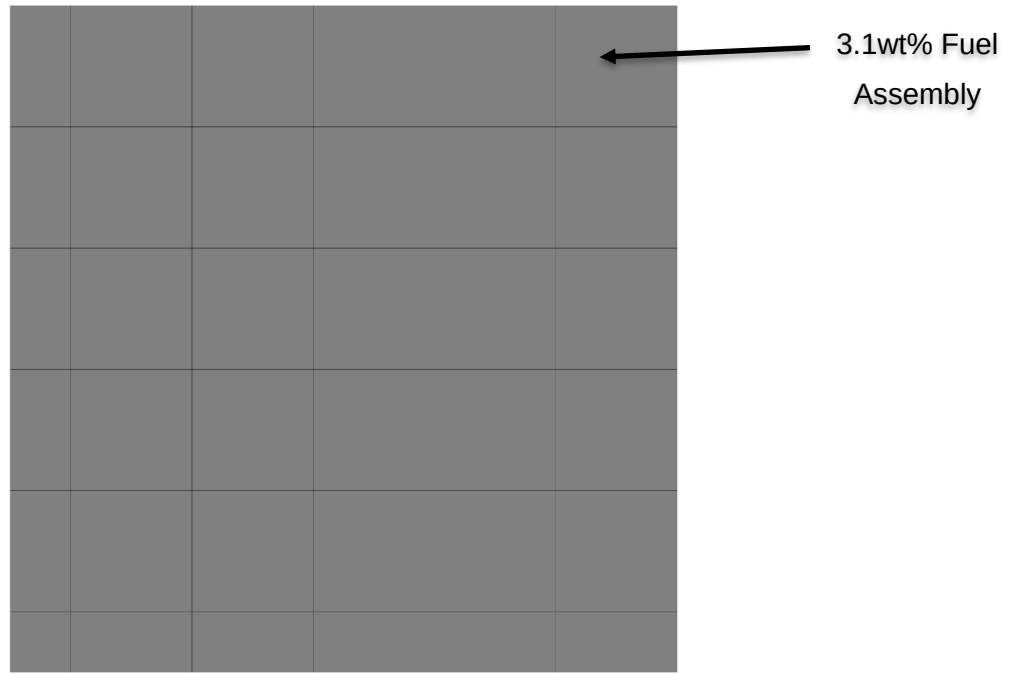


Figure 3-12: Don_1

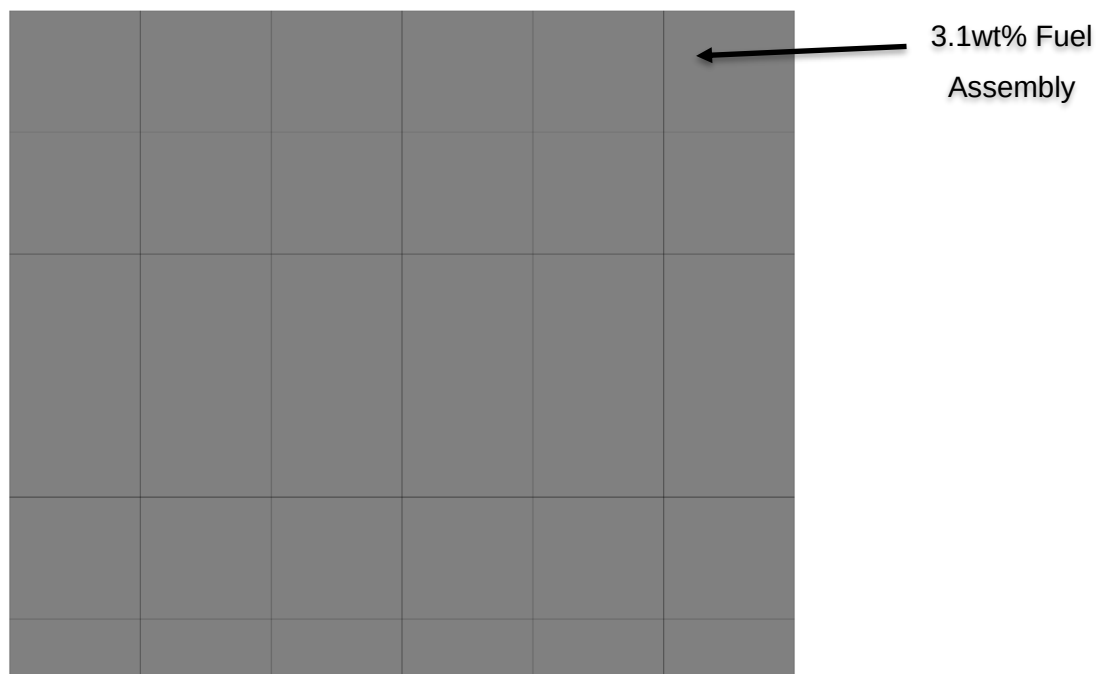


Figure 3-13: Don_2

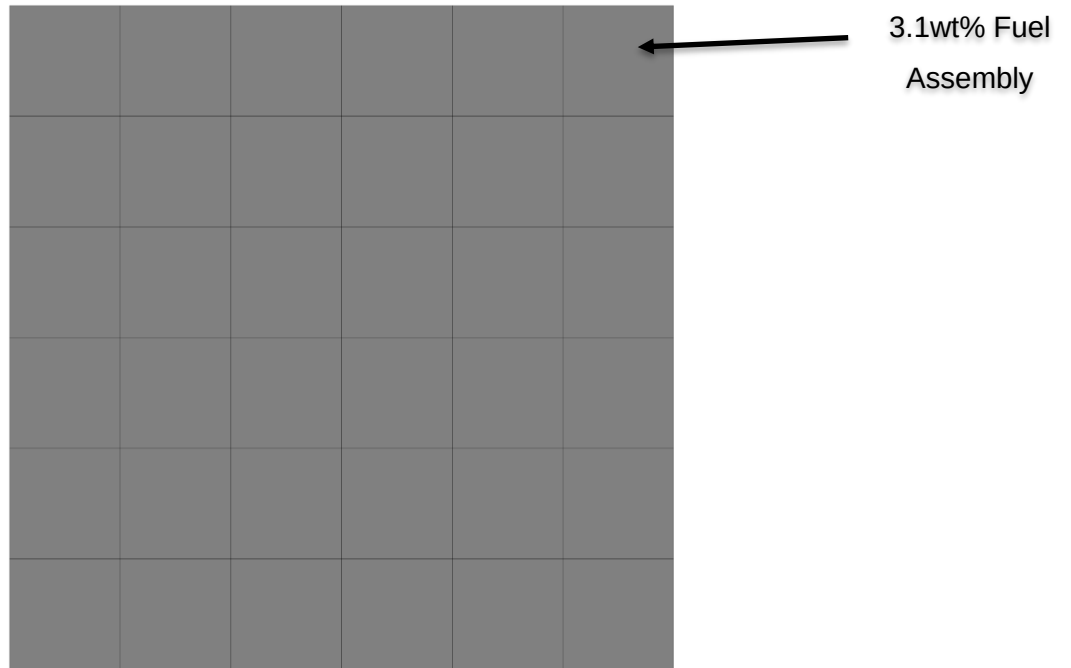


Figure 3-14: Don_3 and Don_4

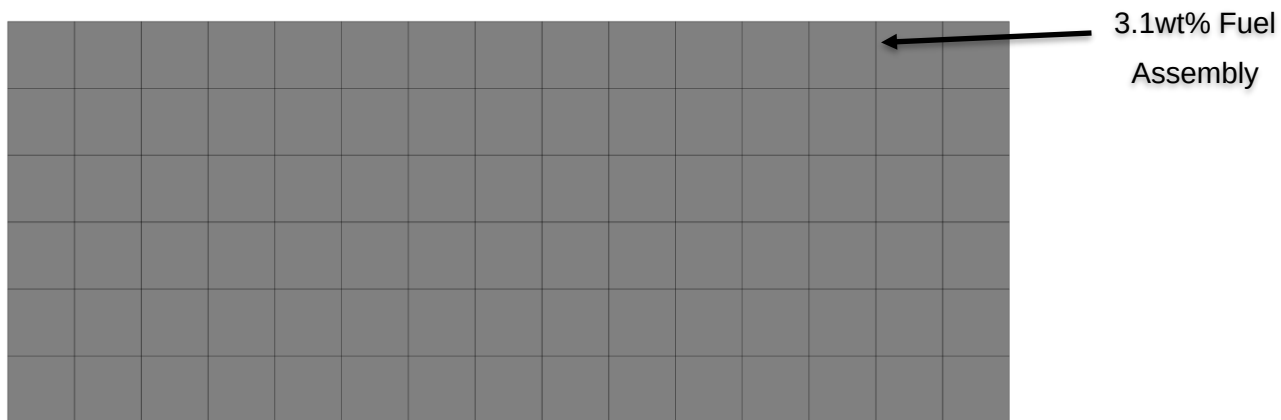


Figure 3-15: Don_5

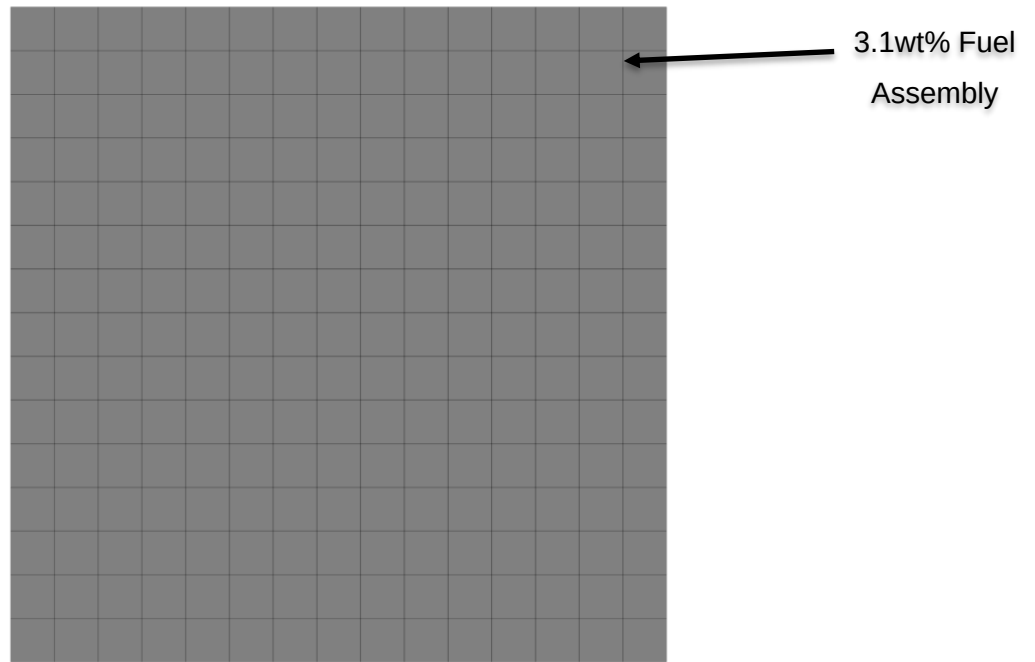


Figure 3-16: Don_6 and Don_7

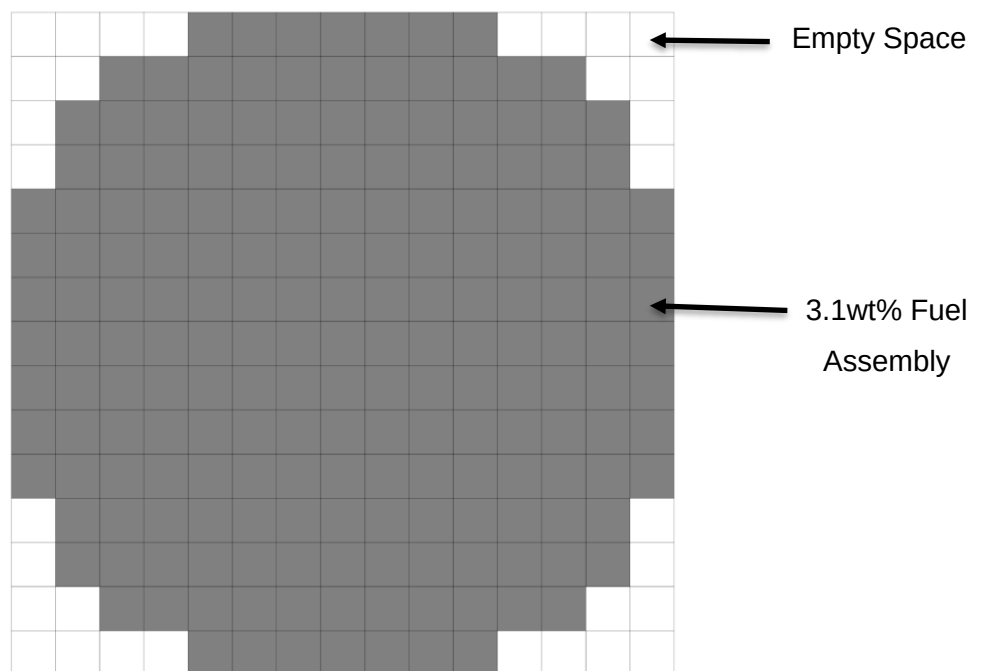


Figure 3-17: Don_8

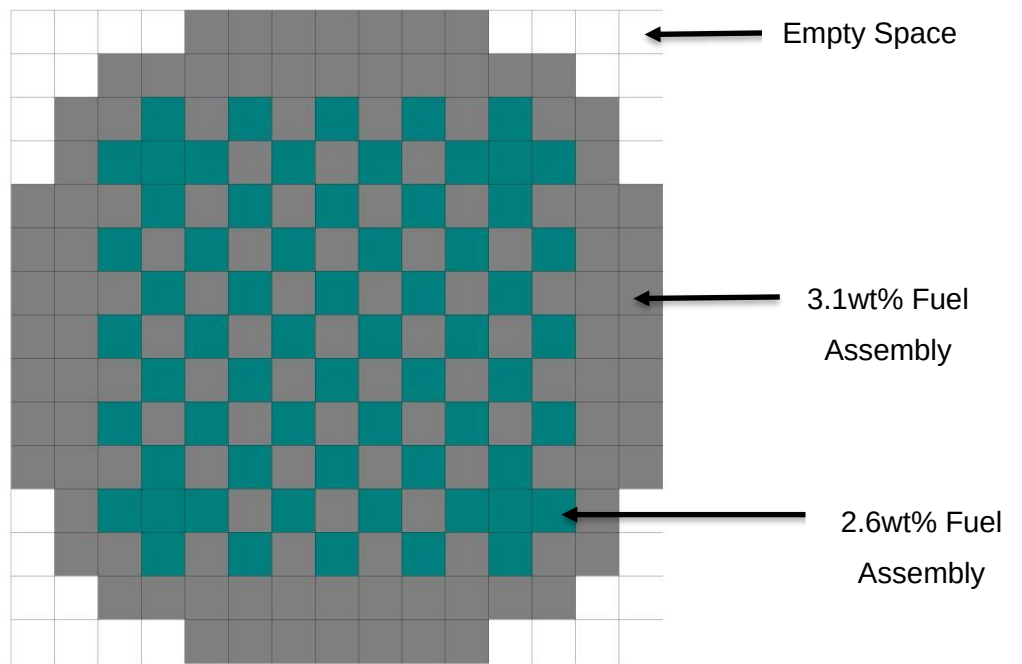


Figure 3-18: Don_9

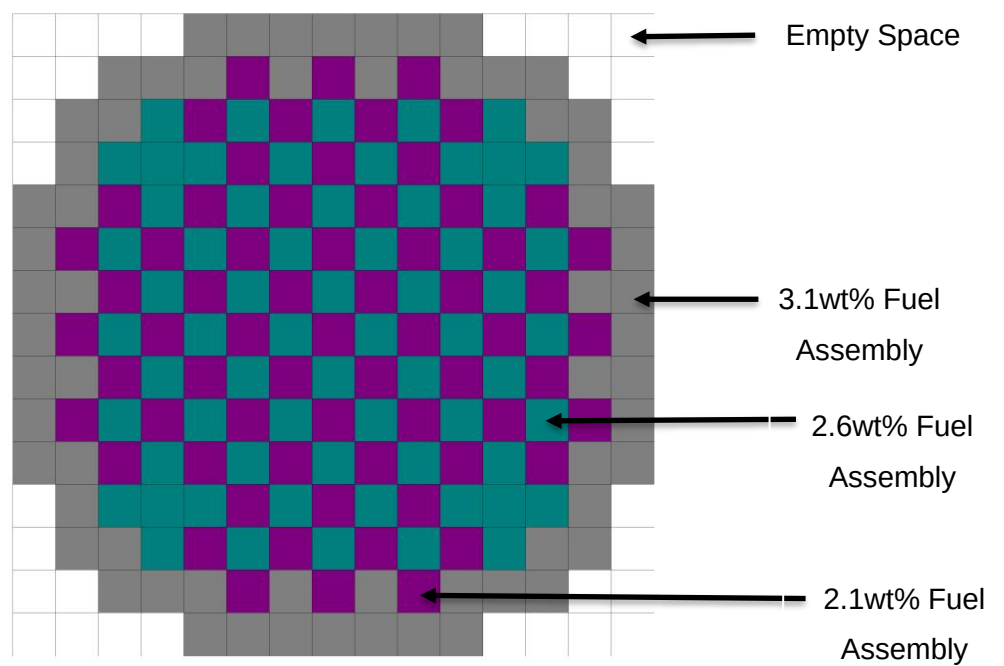


Figure 3-19: Don_10

CHAPTER 4 RESULTS AND DISCUSSION

The bash scripts for compiling and linking the DRAGON and DONJON source codes were written as described in Chapter 3 (Section 3.1). The fuel assembly and full core models of the North Anna reactor were developed through the methodologies also described in Chapter 3 (Sections 3.2 and 3.3). As mentioned in Chapter 3, the assembly models were developed under the assumption that there are no burnable poisons in the assemblies, while the full core models were developed under the assumption that the core is surrounded by empty space instead of moderator.

The assembly models were run in DRAGON while the full core models were run in DONJON. In this chapter the results obtained for the models are presented and discussed. The DRAGON results are presented and discussed in Section 4.1 while the DONJON results are presented and discussed in Section 4.2. The outcomes of running the compiling and linking bash scripts for DRAGON and DONJON are included in Sections 4.1 and 4.2 respectively.

4.1 The DRAGON Results

This section presents and discusses the results which were obtained from running the DRAGON bash scripts and input files described in chapter 3.

4.1.1 Compiling and Linking the DRAGON Source Code

The compiling and linking bash scripts were written according to the methodology described in Section 3.1.1. The compiling bash script was run, and it successfully produced object files from the DRAGON source code. The linking bash script was run, and it successfully linked the object files into a DRAGON executable.

4.1.2 The DRAGON Reference Model: TCWU01-Test Case 4

The DRAGON executable that was obtained was tested by running the reference model TCWU01- Test Case 4. The k_{∞} value that was obtained is presented in Table 4-1 along with its uncertainty. The value of k_{∞} for the reference model as provided by the DRAGON development team is also presented in Table 4-1 for comparison.

Table 4-1: Comparison of the k_{∞} values for TCWU01-Test Case 4

Model	Parameter	Value
TCWU01-Test Case 4	k_{∞} obtained using the obtained DRAGON executable	0.830733
TCWU01-Test Case 4	k_{∞} provided by the DRAGON development team	0.830733

As shown in Table 4-1 above, the value of k_{∞} obtained for the reference model using the DRAGON executable which was produced (Section 4.1.1) is identical to the value of k_{∞} which was provided by the DRAGON development team for the reference model. Therefore, the implementation of the DRAGON executable is reliable since it produced the expected results for the reference model.

It is also noted that the actual value is reported with six significant figures after the decimal point, whereas the first significant figure in the reported uncertainty (0.00000004378) is the eight significant after the decimal point. This effectively means that the error in the multiplication factor is zero when reporting six significant figures after the decimal point. However, the uncertainty value is noted, since it gives an indication of the accuracy between values.

4.1.3 Infinite Multiplication Constants for the DRAGON Models

Table 4-2 below presents the k_{∞} values and uncertainties for all the DRAGON models which were constructed during the modification of the reference model (TCWU01-Test Case 4) into the North Anna fuel assembly models (DRAG_13, DRAG_14 and DRAG_15). A discussion of the k_{∞} values is given below Table 4-2.

As stated and justified in Section 3.2.2, the discussion is limited to the behaviour of the k_{∞} values (increase, decrease or remain unchanged) between consecutive models; possible explanations for the observed behaviour are given, and the observed behaviour is compared to expected behaviour according to literature.

Table 4-2: Infinite Multiplication Constants for the DRAGON Models

Model	k_{∞}
TCWU01-Test Case 4	0.830733
DRAG_0	0.888367
DRAG_1	0.888067
DRAG_2	0.888591
DRAG_3	0.888012
DRAG_4	1.224361
DRAG_5	1.221459
DRAG_6	1.218281
DRAG_7	1.194738
DRAG_8	1.195441
DRAG_9	1.195441
DRAG_10	1.194920
DRAG_11	1.198067
DRAG_12	1.220461
DRAG_13	1.220461
DRAG_14	1.272045
DRAG_15	1.309924

The DRAGON model DRAG_0 was obtained by removing the neutron poison that was dissolved in the moderator of the reference model. The k_{∞} value is expected to increase when the neutron poison concentration decreases (Nichols, 1962). Neutron poisons have a large neutron capture

cross-section (U.S.NRC, 2019) such that reducing the neutron poison concentration leads to fewer neutron capture reactions (less neutrons absorbed) and thus more fission reactions (more neutrons now available for fission). As can be seen from Table 4-2 above, the k_{∞} value does increase from 0.830733 for the reference model to 0.888367 for DRAG_0. Therefore, the DRAGON result for k_{∞} against neutron poison concentration is consistent with basic reactor physics principles and what literature predicts (Nichols, 1962).

From Drag_0, the DRAGON model DRAG_1 was obtained by reducing the fuel cell dimensions from 1.26209cm x 1.26209cm to 1.25894cm x 1.25894cm, thereby reducing the amount of moderator in the fuel cell while the amount of fuel remains constant. For typical pressurized water reactors such as the North Anna reactor, the moderator to fuel ratio is fixed below the optimum value for reactor safety reasons (U.S.DOE, 1993). Therefore, the k_{∞} value for the North Anna reactor is expected to decrease as the moderator to fuel ratio decreases (U.S.DOE, 1993) (Choi & Park, 2017). As reported in Table 4-2, the value of k_{∞} decreases from 0.888367 for DRAG_0 to 0.888067 for DRAG_1. This DRAGON result is therefore consistent with the expected behaviour of k_{∞} in response to reduced moderator to fuel ratio via removing moderator from the fuel cell.

The DRAGON model DRAG_1 was modified into DRAG_2 by increasing the cladding inner and outer radii from 0.39306cm and 0.45802cm respectively to 0.41783cm and 0.47498cm respectively, thereby reducing the cladding thickness and introducing a helium gap in the fuel cell. The actual volume of the cladding per unit length also decreases.

The increase in cladding outer radius means the amount of moderator in the fuel cell is reduced, and therefore the moderator to fuel ratio also decreases such that the k_{∞} value can be expected to decrease, as described in the previous paragraph. However, there are now more factors to consider for example the presence of helium in the fuel cell, and the reduction of cladding thickness.

The presence of helium in the gap is expected to have a negligible effect on the k_{∞} value (Lamarsh & Barrata, 2001). On the other hand, reducing the cladding thickness is expected to allow more thermal neutrons to reach the fuel and more fission neutrons to reach the moderator than with a thicker cladding due to reduced neutron attenuation (Lamarsh & Barrata, 2001), thereby increasing the k_{∞} value. The resultant behaviour of the k_{∞} value when DRAG_2 is obtained from DRAG_1 will thus depend on the combined effect of reduction in moderator to fuel ratio and reduction in cladding thickness. From Table 4-2 the value of k_{∞} increases from 0.888067 for DRAG_1 to 0.888591 for DRAG_2. This result means that the effect of reducing cladding thickness dominated the effect of reducing the moderator to fuel ratio.

Going from model DRAG_2 to model DRAG_3, the fuel pellet size was increased from 0.39306 cm to 0.409578cm, thus increasing the amount of fuel in the fuel cell while retaining the fuel enrichment. One therefore expects that k_{∞} will increase with increasing fissile content (U.S.DOE, 1993).

However, since the amount of moderator remains unchanged, the moderator to fuel ratio is decreased when the fuel pellet size is increased. The value of k_{∞} is then expected to decrease as the moderator to fuel ratio decreases (U.S.DOE, 1993). As reported in Table 4-2, the value of k_{∞} decreases from 0.888591 for DRAG_2 to 0.888012 for DRAG_3. This DRAGON result therefore means that the effect of the moderator to fuel ratio is stronger than the increase in fissile content in terms of the multiplication constant.

Going from model DRAG_3 to model DRAG_4, the fuel enrichment was increased from 0.711w/o to 2.1w/o. Increasing the enrichment of Uranium fuel increases the number of U-235 (fissile) atoms and thus the fission macroscopic cross-section, such that the number of fission reactions in the fuel cell increases (Lamarsh & Barrata, 2001). Thus, the value of k_{∞} is expected to increase as the fuel enrichment increases. Table 4-2 reports that the value of k_{∞} increases from 0.888012 for DRAG_3 to 1.224361 for DRAG_4. Therefore, this DRAGON result agrees with the expected behaviour of k_{∞} when fuel enrichment is increased.

From model DRAG_4 to model DRAG_5, the moderator number density was reduced from $0.0221163 \times 10^{24} / \text{cm}^3$ to $0.021650 \times 10^{24} / \text{cm}^3$. Since the volume of the moderator is unchanged, then the number of moderator particles is reduced when the moderator density is reduced. Thus, the moderator to fuel ratio decreases, such that the value of k_{∞} is expected to decrease (U.S.DOE, 1993). Table 4-2 shows that the value of k_{∞} decreases from 1.224361 for DRAG_4 to 1.221459 for DRAG_5. This DRAGON result therefore agrees with the expected response of k_{∞} to a decrease in moderator to fuel ratio due to a reduction in moderator density.

Going from model DRAG_5 to DRAG_6, the number density of the cladding was changed from $3.83243 \times 10^{22} \text{ molecules/cm}^3$ to $4.25 \times 10^{22} \text{ molecules/cm}^3$. As shown in Table 4-2 the value of k_{∞} decreases from 1.221459 for DRAG_5 to 1.218281 for DRAG_6. This can be explained by the significant increase in the number density of the cladding. Since there are now significantly more particles in the cladding than before, then the total macroscopic cross-section of the cladding is now significantly larger (Lamarsh & Barrata, 2001). Thus, the rate of reaction between neutrons and the cladding is significantly higher.

Therefore, in DRAG_6, neutrons produced in the fuel cannot reach the moderator as much as they can in DRAG_5. That reduces the number of neutrons in the moderator. Moreover, the thermalized neutrons from the moderator cannot enter the fuel rods in DRAG_6 as much as they could in DRAG_5. The rate of fission will therefore be lower in DRAG_6 than in DRAG_5, which what is observed from Table 4-2. The DRAGON result therefore agrees with reactor physics theory.

From model DRAG_6 to model DRAG_7, the fuel temperature was increased from 600 K to 1200 K. This results in Doppler broadening of the resonances of the fuel isotopes (Lamarsh & Barrata, 2001). The resulting change in the reactivity (or multiplication factor) allows the definition of the fuel temperature coefficient. For safety reasons, all reactors are designed to have negative fuel temperature coefficients (Lamarsh & Barrata, 2001). That is, the value of k_{∞} is expected to decrease as fuel temperature increases. According to Table 4-2 the value of k_{∞} decreases from 1.218281 for DRAG_6 to 1.194738 for DRAG_7. This DRAGON result thus agrees with the expected behaviour of k_{∞} when fuel temperature increases.

Going from model DRAG_7 to model DRAG_8, the moderator temperature was decreased from 600 K to 550 K. Like with the fuel temperature reactivity mentioned in the previous paragraph, reactors are also designed to have negative moderator temperature reactivity coefficient for reactor safety (Lamarsh & Barrata, 2001). That is, the k_{∞} value is expected to increase as the moderator temperature is reduced. As shown in Table 4-2, the value of k_{∞} increases from 1.194738 for DRAG_7 to 1.195441 for DRAG_8. Therefore, this DRAGON result is consistent with the expected behaviour of k_{∞} when the moderator temperature is decreased.

From model DRAG_8 to model DRAG_9, the cladding temperature was increased from 600K to 900K. As shown in Table 4-2, the value of k_{∞} did not change in response to the increase in cladding temperature. This indicates that the isotopes defined for the cladding did not have any self-shielding treatment in terms of Doppler broadening when the cross-section libraries were prepared. Furthermore, this result is against the intuition that the higher cladding temperature would increase the moderator temperature and thus reduce the value of k_{∞} (Lamarsh & Barrata, 2001). While not intuitive, it is expected since the DRAGON calculations carried out in this research project were not coupled neutronics and thermal-hydraulics calculations and as such, the increased cladding temperature does not automatically increase the moderator temperature. The moderator temperature remains constant until it is manually changed by the user.

The fuel cell of model DRAG_9 was used to form a 17x17 fuel assembly to obtain model DRAG_10. As shown in Table 4-2, the k_{∞} value decreases from 1.195441 for DRAG_9 to 1.194920 for DRAG_10 with a difference of 0.000921. It is expected that both models should have

identical values, since both are infinite systems of the same fuel pin and have reflective boundary conditions. The DRAGON result from DRAG_9 to DRAG_10 is therefore not what is expected. Further investigation is required to understand why DRAGON produces this result.

From model DRAG_10 to model DRAG_11, a layer of moderator 0.0508 cm thick was added on each side of the fuel assembly. The assembly moderator to fuel ratio was thus increased. As already mentioned in the previous paragraphs, the value of k_{∞} is expected to increase with moderator to fuel ratio (U.S.DOE, 1993). Table 4-2 shows that the value of k_{∞} does increase from 1.194920 for DRAG_10 to 1.198067 for DRAG_11. This DRAGON result thus agrees with the expected k_{∞} behaviour when the moderator to fuel ratio is increased via increasing the moderator.

Model DRAG_12 was obtained by replacing 25 of the fuel pins in DRAG_11 with guide tubes and an instrumentation tube. Thus, fuel was removed from the fuel assembly and replaced with moderator. The moderator to fuel ratio of the assembly was therefore increased. With an increase in moderator to fuel ratio, the value of k_{∞} is expected to increase (U.S.DOE, 1993). As shown in Table 4-2, the value of k_{∞} does increase from 1.198067 for DRAG_11 to 1.220461 for DRAG_12. Therefore, this DRAGON result is consistent with the expected response of k_{∞} to introduction of guide thimbles in the fuel assembly.

Models DRAG_12 and DRAG_13 are identical except for some calculational specifications which were changed for the purpose of creating the DONJON inputs. Their k_{∞} value is thus identical, 1.220461, as shown in Table 4-2.

From model DRAG_13 to model DRAG_14, the fuel enrichment was increased from 2.1 w/o to 2.6 w/o. Increasing fuel enrichment increases the number of fissile atoms, in turn increasing the fission macroscopic cross-section and thus the number of fission reactions (Lamarsh & Barrata, 2001). Thus, the value of k_{∞} is expected to increase with fuel enrichment. As shown in Table 4-2, the value of k_{∞} increases from 1.220461 for DRAG_13 to 1.272045 for DRAG_14. This DRAGON result therefore agrees with the expected response of k_{∞} to increase in fuel enrichment.

From model DRAG_14 to model DRAG_15, the fuel enrichment was increased from 2.6 w/o to 3.1 w/o. The argument for this modification is like the one made in the previous paragraph. Increasing fuel enrichment increases the number of fissile atoms, in turn increasing the fission macroscopic cross-section and thus the number of fission reactions (Lamarsh & Barrata, 2001). Thus, the value of k_{∞} is expected to increase with fuel enrichment. As shown in Table 4-2, the value of k_{∞} increases from 1.272045 for DRAG_14 to 1.309924 for DRAG_15. Therefore, this

DRAGON result is consistent with the expected behaviour of k_{∞} when fuel enrichment is increased.

4.1.4 The 2-Group Constants for the North Anna Assembly Models

The 2-group DRAGON results obtained for the North Anna assembly models are presented in Table 4-3. The discussion of each of the 2-group constants is given after Table 4-3.

Table 4-3: The 2-group Constants for the Final Fuel Assembly Models

Models	$\Sigma_a(10^{-2}/cm)$		$\nu\Sigma_f(10^{-2}/cm)$		$\Sigma_s(10^{-2}/cm)$		χ		SPH	
	g1	g2	g1	g2	g1	g2	g1	g2	g1	g2
DRAG_13	0.92959	6.4363	0.56552	10.234	48.5286	117.070	1	0	1	1
DRAG_14	0.96656	7.3643	0.63834	12.246	48.4433	116.240	1	0	1	1
DRAG_15	0.98964	8.2430	0.70901	14.147	48.3581	115.777	1	0	1	1

χ

The quantity χ is defined as the fraction of fission neutrons which are produced in each energy group. As shown in Table 4-3, χ equals 1 in group 1 for all three fuel assembly models. This is consistent with reactor physics theory, which states that fission neutrons are produced with fast energies (Lamarsh & Barrata, 2001). Since fast energies belong to group 1, then all fission neutrons released belong to group1.

SPH

The SPH factor is used by DRAGON to check if the equivalence principle (see Section 2.1) is satisfied during the group-collapse and homogenization calculations. It is the ratio of the reaction rates of the heterogeneous and the homogenized assembly models. If the ratio is 1, then the equivalence requirement is satisfied. As shown in Table 4-3, the SPH factor for all the assembly models in all energy groups is equal to 1. Therefore, the equivalence requirement was satisfied for all the models in both energy groups.

$$\nu\Sigma_f$$

Two main observations can be made from the values of $\nu\Sigma_f$ in Table 4-3. The first observation is that in both group 1 and group 2, the values of $\nu\Sigma_f$ increase with fuel enrichment (i.e. from DRAG_13 to DRAG_15). The second observation is that for all three assembly models, the values of $\nu\Sigma_f$ are larger in group 2 than in group 1.

- For the first observation:
Since ν is constant in each group (Stacey, 2001), the observation must be due to Σ_f . As fuel enrichment increases, the number density of U-235 increases. Since Σ_f is the product of number density and microscopic fission cross-section, then Σ_f also increases with fuel enrichment. Hence $\nu\Sigma_f$ increases with fuel enrichment.
- For the second observation:
The microscopic fission cross-section of U-235 is larger in group 2 than in group 1 (Stacey, 2001). So, for the same fuel enrichment and thus number density of U-235, the value of Σ_f is larger in group 2 than in group 1. Moreover, ν is 2.432 in group 2 and 2.349 in group 1 (Stacey, 2001). Therefore, $\nu\Sigma_f$ is larger in group 2 than in group 1 for each enrichment.

$$\Sigma_a$$

For each assembly, the macroscopic absorption cross-section is the sum of the macroscopic fission cross-section (Σ_f) and the macroscopic neutron capture cross-section (Σ_γ) (Stacey, 2001).

Two main observations about Σ_a can be made from Table 4-3. Firstly, for group 1 and group 2, Σ_a increases with fuel enrichment. Secondly, for all three assembly models, Σ_a is larger in group 2 than in group 1. These two observations are identical to the observations made for Σ_f above.

Therefore, the behaviour of Σ_a is determined by the behaviour of Σ_f in both energy groups as fuel enrichment increases.

Σ_s

Table 4-3 shows that for both energy groups, the macroscopic scattering cross-section decreases with fuel enrichment. Table 4-3 further shows that for each assembly model, the macroscopic scattering cross-section is larger in group 2 than it is in group 1.

- For the First Observation:
U-238 has a larger microscopic scattering cross-section than U-235 (Stacey, 2001). As the fuel enrichment increases, the number density of U-235 increases while the number density of U-238 decreases. The result is that the macroscopic scattering cross-section U-238 contributes less to the total macroscopic scattering cross-section of the fuel as enrichment increases. On the other hand, the macroscopic scattering cross-section of U-235 contributes more to the total macroscopic scattering cross-section of the fuel as enrichment increases.
Ultimately, the smaller scattering cross-section of U-235 replaces the larger scattering cross-section of U-238 as fuel enrichment increases. Therefore, the total macroscopic scattering cross-section of the fuel decreases as fuel enrichment increases.
- For the Second Observation:
The microscopic scattering cross-section of both U-235 and of U-238 are larger in group 2 than they are in group 1. Therefore, the fuel macroscopic scattering cross-section is also larger in group 2 than in group 1 for all three assembly models.

4.2 The DONJON Results

In this section, the results obtained from running the DONJON bash scripts and input files described in Chapter 3 are presented and discussed.

4.2.1 Compiling and Linking the DONJON Source Code

The compiling and linking bash scripts were written according to the methodology described in Section 3.1.2. The compiling bash script was run, and it successfully produced object files from the DONJON source code. Afterwards, the linking bash script was run, and it successfully linked the object files into a DONJON executable.

4.2.2 The DONJON Reference Model: LMW43D

The DONJON executable that was obtained was tested by running the reference model LMW43D. The k_{eff} value that was obtained is presented in Table 4-5. The value of k_{eff} for the reference model as provided by the DONJON development team is also presented in Table 4-5 for comparison.

Table 4-4: Comparison of k_{eff} for the Reference Model LMW43D

Model	Parameter	Value
LMW43D	k_{eff} obtained using the obtained DONJON executable	1.00522
LMW43D	k_{eff} provided by the DONJON development team	1.00522

As shown in Table 4-4 above, the value of k_{eff} obtained for the reference model using the DONJON executable (Section 4.2.1) is identical to the value of k_{eff} which was provided by the DONJON development team for the reference model. The implementation of the DONJON executable is thus reliable since it produced the expected results for the reference model.

4.2.3 Effective Multiplication Constants for the DONJON Models

Table 4-5 below presents the k_{eff} values for all the DONJON models which were constructed during the modification of the reference model LMW43D into the North Anna full core model (DON_10). A discussion of the k_{eff} values is given below Table 4-5.

As stated and justified in Section 3.2.2, the discussion is limited to the response of the k_{eff} values (increase, decrease or remain unchanged) between consecutive models. Possible explanations for the observed behaviour are given, and the observed response is compared to expected behaviour according to literature.

Table 4-5: Effective Multiplication Constants for the DONJON Models

Models	k_{eff}
LMW43D	1.006484
DON_1	1.22052
DON_2	1.27746
DON_3	1.280424
DON_4	1.291475
DON_5	1.29820
DON_6	1.30499
DON_7	1.30322
DON_8	1.29484
DON_9	1.27621
DON_10	1.23559

The DONJON model DON_1 was obtained by replacing the CRE: module with the MACD: module and replacing the specified material with the 3.1 w/o DRAGON fuel assembly output. The boundary conditions were also changed to void. The value of k_{eff} increased from 1.00522 for the reference model to 1.22052 for DON_1, as shown in Table 4-5. Since no information about the fuel in the reference model (type, temperature, enrichment) is given, a comparison with the 3.1 w/o DRAGON output cannot be given.

The modifications from DON_1 to DON_8 all involve changing the size of the reactor core. Since the DONJON reactor models were specified with void boundary conditions, then they are finite reactor cores which experience neutron leakage (Lamarsh & Barrata, 2001).

For finite reactors, the amount of neutron leakage can be reduced by increasing the size of the reactor core (Gupta, 1989). By increasing the size of the reactor core, the probability of fission neutrons escaping the core decreases, and thus neutron leakage decreases. A reduction in neutron leakage leads to increased neutron flux in the core and thus increased number of fissions, which in turn leads to an increase in the value of k_{eff} (Lamarsh & Barrata, 2001). Conversely, reducing the reactor core size leads to increased neutron leakage (Gupta, 1989) and thus a decrease in the value of k_{eff} (Lamarsh & Barrata, 2001).

From model DON_1 to model DON_2, the x-dimensions of all the fuel assemblies in the core were increased to 21.50358 cm. This was to show the change only in terms of the x-dimension. As shown in Table 4-5, the value of k_{eff} increased from 1.22052 for DON_1 to 1.27746 for DON_2.

Going from model DON_2 to MODEL DON_3, the y-dimensions of all the fuel assemblies in the core were next increased to 21.50358 cm. Table 4-5 shows that the value of k_{eff} increased from 1.27746 for DON_2 to 1.28042 for DON_3.

From model DON_3 to model DON_4, the z-dimensions of all the assemblies in the core were increased to 24.332 cm; the z-dimensions were as noted in Table 3-4. Table 4-5 shows that the value of k_{eff} increased from 1.28042 for DON_3 to 1.29148 for DON_4.

Going from model DON_4 to model DON_5, the number of fuel assemblies along the x-axis were increased from 6 to 15. This modification resulted in the value of k_{eff} increasing from 1.29148 for DON_4 to 1.29820 for DON_5, as shown in Table 4-5.

From model DON_5 to model DON_6, the number of fuel assemblies along the y-axis were increased from 6 to 15. As shown in Table 4-5, the value of k_{eff} increased from 1.29820 for DON_5 to 1.30499 for DON_6.

Going from model DON_6 to model DON_7, the number of fuel assemblies along the z-axis were decreased from 20 to 15. Table 4-5 shows that the value of k_{eff} decreased from 1.30499 for DON_6 to 1.30322 for DON_7.

From model DON_7 to model DON_8, the assemblies at the corners of the core were replaced with empty space. As shown in Table 4-5, the value of k_{eff} decreased from 1.30322 for DON_7 to 1.29484 for DON_8.

As noted in the above paragraphs, the value of k_{eff} increased for the modifications that were made from DON_1 to DON_6 (increasing core size), while the value of k_{eff} decreased for the modifications which were made from DON_6 to DON_8 (decreasing core size). These DONJON results are therefore consistent with the expected response of the value of k_{eff} when the reactor core size is increased or decreased.

Going from model DON_8 to model DON_9, some of the 3.1w/o assemblies in the core were replaced with 2.6w/o assemblies. As a result, the value of k_{eff} decreased from 1.29484 for DON_8 to 1.27621 for DON_9 as shown in Table 4-5.

From model DON_9 to model DON_10, some of the remaining 3.1w/o assemblies in the core were replaced with 2.1w/o assemblies. Table 4-5 shows that this modification caused the value of k_{eff} to decrease from 1.27621 for DON_9 to 1.23559 for DON_10.

When the higher enrichment assemblies are replaced by the lower enrichment assemblies from model DON_8 to model DON_10, the number density of fissile atoms in the reactor core decreases. Thus, the macroscopic fission cross-section of the core decreases. As a result, the rate of fission reactions in the core decreases, leading to a decrease in the value of k_{eff} (Lamarsh & Barrata, 2001).

As mentioned within the previous three paragraphs, the value of k_{eff} was found to decrease when some of the 3.1w/o fuel assemblies were replaced with 2.6w/o assemblies for DON_9 and with both 2.6w/o and 2.1w/o assemblies for DON_10. Therefore, these DONJON results agree with the expected behaviour of the k_{eff} value when higher enrichment assemblies are replaced with lower enrichment assemblies in a reactor core.

4.2.4 Full Core (DON_10) Flux and Power Results

The neutron flux values for the North Anna full core model (DON_10) were obtained. The flux values were specified at each fuel assembly in the core. For each assembly, the flux was specified as two values, one in energy group 1 and the other in energy group 2. The corresponding power values for each fuel assembly of volume V were calculated using the formula: $Power = V \times E_f \times \Sigma_f \times \phi$ (Lamarsh & Baratta, 2001).

The full core flux and power profiles are presented and discussed in this section. Since the core model is three dimensional, these profiles are presented in both the axial direction (along the core's height) and radial direction (along the core's xy plane).

4.2.4.1 The Neutron Flux Profiles

Tables 4-6 and 4-7 below present the 2-group radial and axial flux results respectively. The radial fluxes of the top plane (Plane 1) and the middle plane (Plane 8) of the reactor core in Table 4-6 illustrate how the radial flux profile changes at these two radial positions along the height of the core. For each of the two planes, the radial flux profile is represented by the flux due to the 15 fuel assemblies along the middle of the plane, as highlighted in yellow dashed lines in Figure 4-1 below.

The axial flux due to fuel assemblies along the centre of the core and the axial flux due to fuel assemblies near the edge of the core in Table 4-7 illustrate how the axial flux profile varies along the axial directions of the core. Assuming Figure 4-1 below is the top plane of the core, then the assemblies along the centre of the core would be below the assembly marked "C", while the assemblies near the edge of the core would be below the assembly marked "E". The assemblies near the core's edge are considered instead of the assemblies at the core's edge so that the comparison of the axial flux profiles can be between fuel assemblies of the same enrichment.

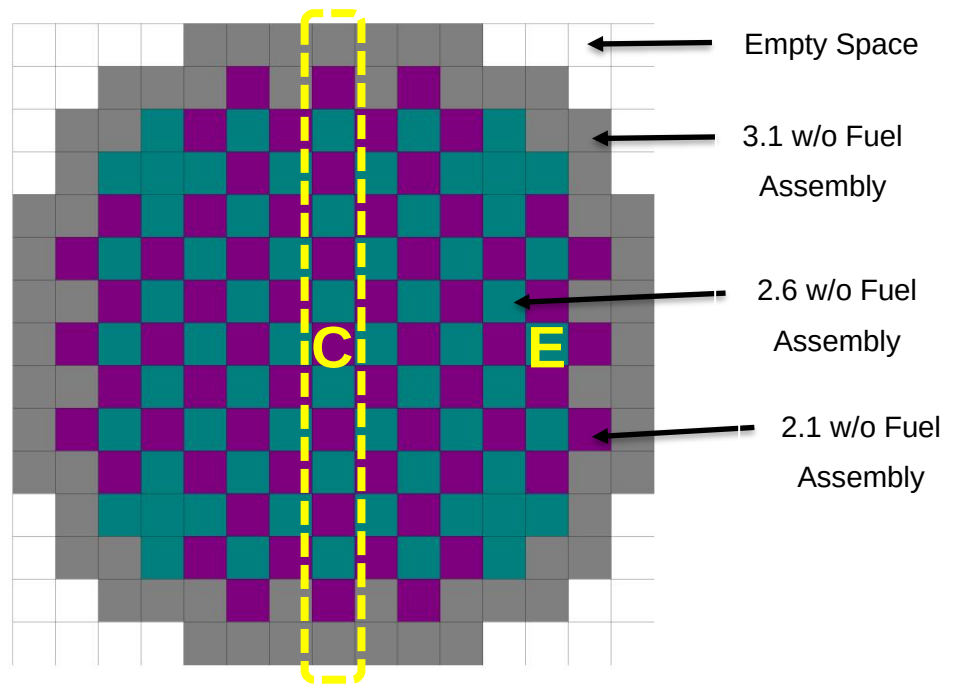


Figure 4-1: Representative Assemblies for the Radial Flux Profile of Each Plane

Table 4-6: Radial Neutron Flux along the Middle of Planes 1 and 8

Fuel Assemblies	Plane 1 flux (10^{13} neutrons/cm ²)		Plane 8 flux (10^{13} neutrons/cm ²)	
	Group 1	Group 2	Group 1	Group 2
1	0.661	0.106	5.49	0.884
2	1.38	0.295	11.5	2.45
3	2.08	0.383	17.3	3.19
4	2.44	0.523	20.3	4.35
5	3.02	0.557	25.1	4.63
6	3.14	0.671	26.1	5.58
7	3.53	0.650	29.3	5.41
8	3.38	0.723	28.1	6.02
9	3.53	0.650	29.3	5.41
10	3.14	0.671	26.1	5.58
11	3.02	0.557	25.1	4.63
12	2.44	0.523	20.3	4.35
13	2.08	0.383	17.3	3.19
14	1.38	0.295	11.5	2.45
15	0.661	0.106	5.49	0.884

The graphical representations of the fluxes in Table 4-6 are given in Figure 4-2 below. The following observations can be made from Figure 4-2:

- I. The radial flux profiles have a non-smooth appearance.
- II. For each plane, the assemblies have larger fluxes in group 1 than in group 2.
- III. In both energy groups, assemblies in plane 8 have higher fluxes than assemblies in Plane1.

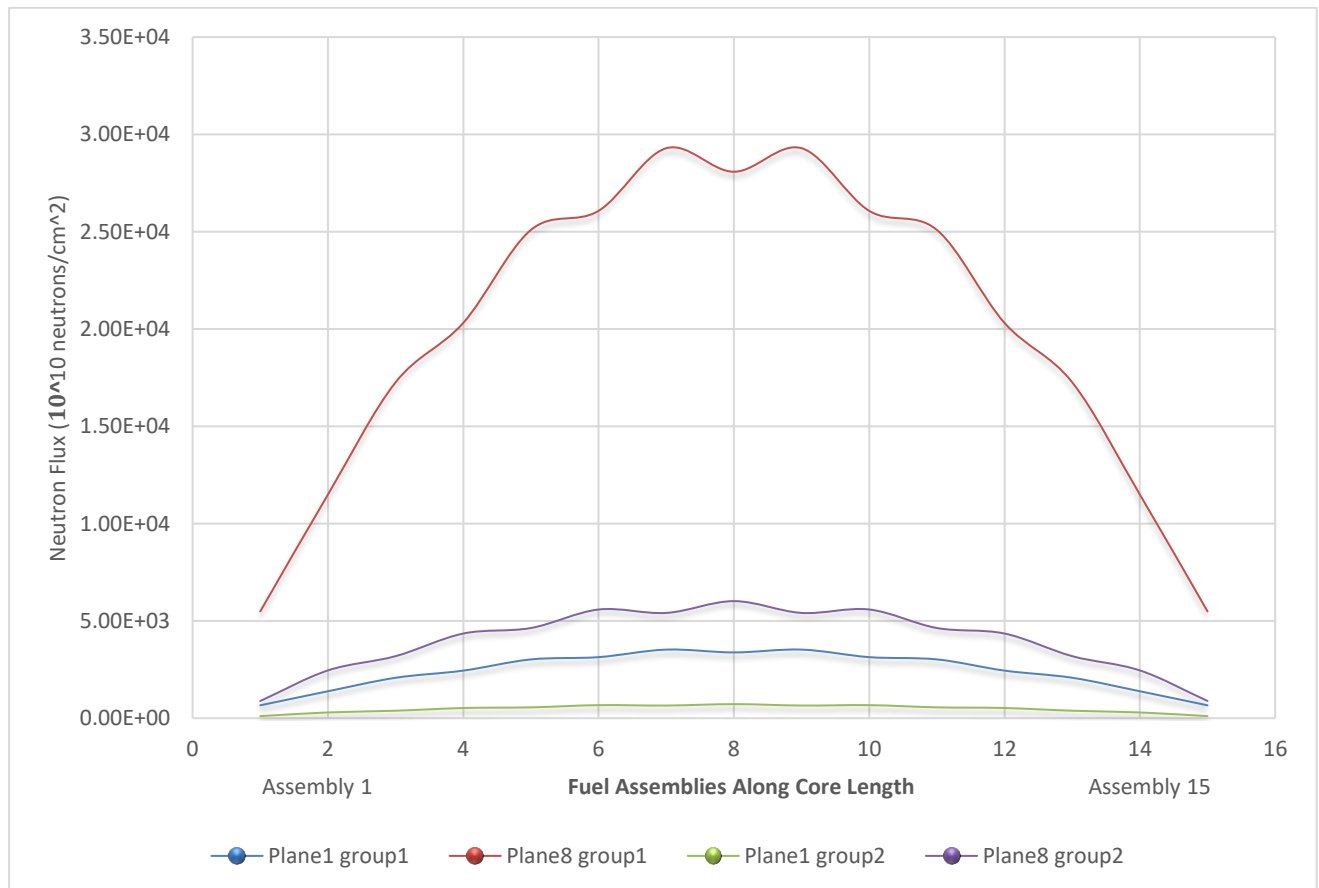


Figure 4-2: The 2-group Radial Neutron Flux Profiles for Planes 1 and 8

I. The Radial Flux Profiles Have A Non-Smooth Appearance

The 2.1 w/o and 2.6 w/o fuel assemblies are arranged in an alternating pattern, as shown in Figure 4-1. The 2.6 w/o assembly has a larger macroscopic fission cross-section than the 2.1 w/o assembly. As a result, there are more thermal neutrons absorbed in the 2.6 w/o assembly than in the 2.1w/o assembly (Lamarsh & Barrata, 2001), and thus more fission reactions in the 2.6 w/o assembly. The higher number of fission reactions in the 2.6 w/o assembly also means more fast neutrons are produced than in the 2.1 w/o assembly.

Consider for instance the profile “Plane 8 group 1” in Figure 4-2. As predicted in the previous paragraph, the 2.1 w/o assemblies must have a lower fast flux profile than the 2.6 w/o assemblies. The 2.1 w/o assemblies are the 2nd, 4th, 6th, 8th, 10th, 12th and 14th assemblies in Figure 4-2. Thus, joining these assemblies should form the 2.1 w/o fast flux profile, shown as the blue dashed line in Figure 4-3. Similarly, joining the 3rd, 5th, 7th, 9th, 11th, and 13th assemblies should form the 2.6 w/o fast flux profile, shown as the orange dashed line in Figure 4-3. As predicted, the 2.1 w/o fast flux profile is indeed lower than the 2.6 w/o fast flux profile.

The same analysis as in the previous paragraph can be done for the other flux profiles in Figure 4-2 to show that the bumpy appearance of the profiles is indeed due to a combination of two profiles, one for the 2.1 w/o assemblies and the other for the 2.6 w/o.

The thermal (group 2) flux profile of the 2.1w/o assembly must be higher than the thermal flux of the 2.6w/o assembly, while the fast (group 1) flux profile of the 2.1w/o assembly must be lower than the fast flux profile of the 2.6w/o assembly. This prediction works well in explaining the bumpy appearance of the radial profiles in Figure 4-2. The profiles are bumpy because each of them is a combination of two profiles, one for the 2.1w/o assemblies and the other for the 2.6w/o assemblies.

When one considers the thermal flux, one might think that a similar form of the flux would appear, i.e. because the higher enriched fuel assembly produces a larger fast flux, and the fast flux slows down to the thermal flux, a higher thermal flux for the higher enriched fuel assembly is expected. However, this is not the case. The opposite is observed.

A possible reason for this is that as fuel enrichment increases, the rate of thermal fissions also increases since more U-235 atoms are available to absorb the thermal neutrons and undergo fission. This could therefore lead to the relative decrease in the thermal flux with increasing enrichment.

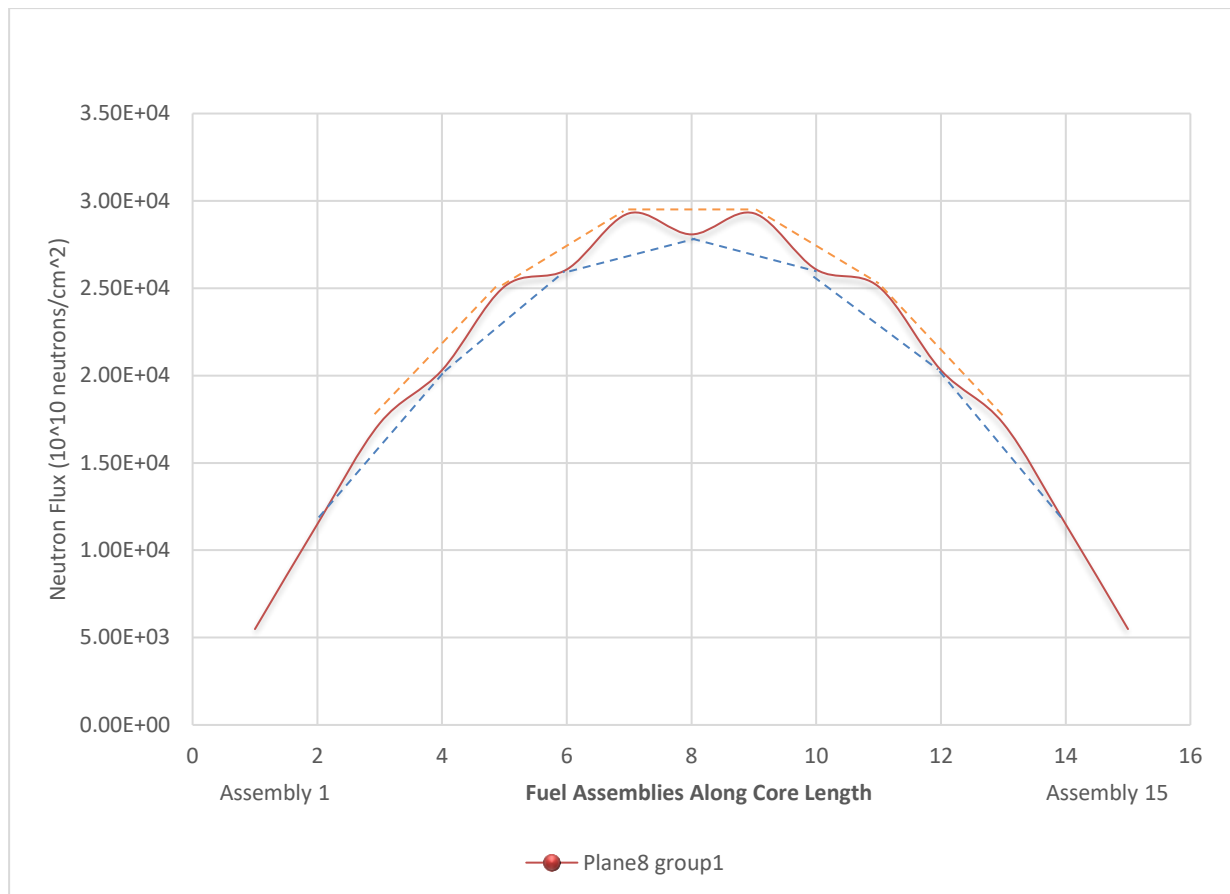


Figure 4-3: “Plane8 group1” Profile Decomposed into 2.1w/o (blue dashed) and 2.6w/o (orange dashed) Profiles.

II. In Each Plane, Assemblies Have Larger Fluxes in Group 1 Than in Group 2

The neutron flux in group 1 is due to the fast neutrons which are produced during fission. When these fast neutrons interact with the reactor materials, some of them are scattered until they enter energy group 2. However, some are captured in the resonance region of U-238. Thus, the fast neutron flux is larger than the thermal neutron flux. Therefore, the DONJON flux results are consistent with physical arguments as far as observation (II) is concerned, as outlined in the above paragraph.

III. In Each Energy Group, Assemblies in Plane 8 Have Larger Fluxes Than Assemblies in Plane 1

Planes 1 and 8 are a part of the set of planes in the axial direction, and this observation is a description of how the flux behaves axially and this is discussed next.

The graphical representation of the axial flux results in Table 4-7 are presented in Figure 4-4. From the axial flux profiles in Figure 4-4, the following is noted:

- I. The axial profiles are smooth.
- II. The assembly fluxes are larger in group 1 than in group 2.
- III. The assemblies along the core's centre have larger fluxes than the assemblies near the core's edge.

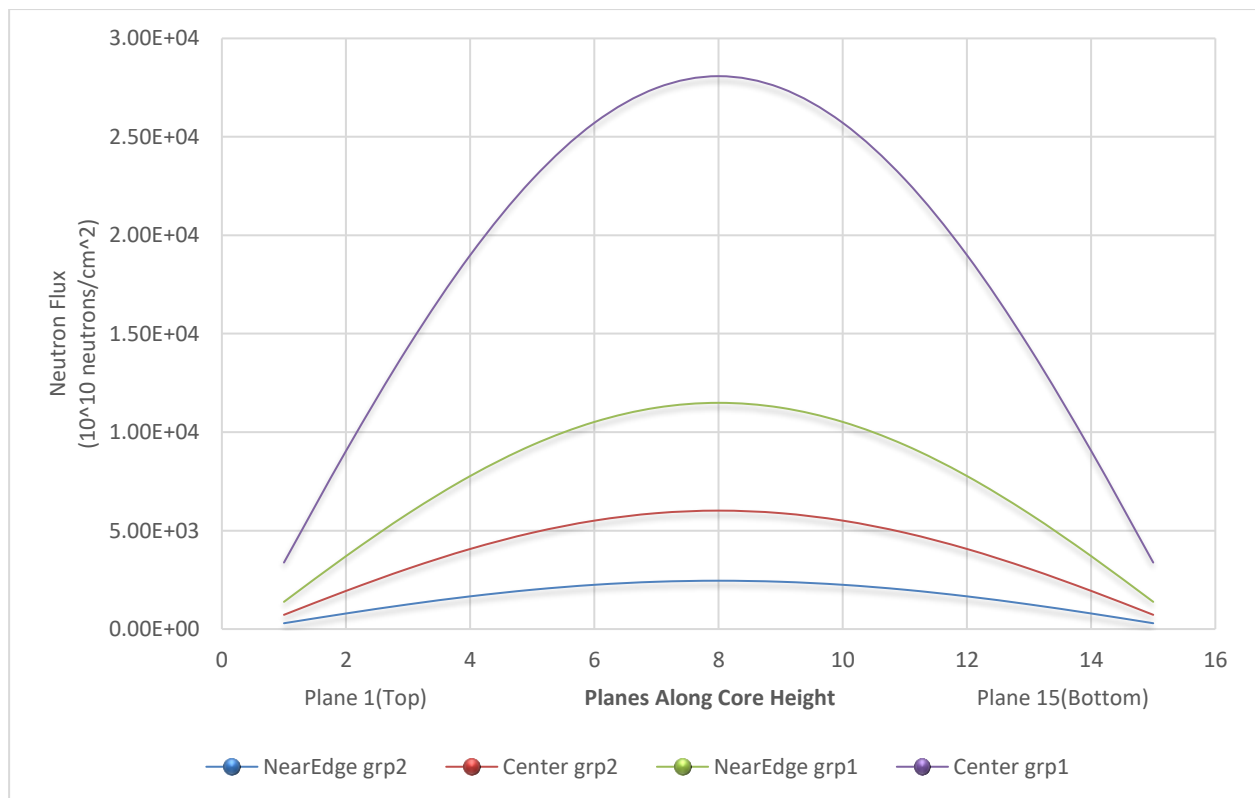


Figure 4-4: The 2-group Axial Neutron Flux Profiles along the Core's Centre and Edge

Table 4-7: Axial Neutron Flux along the Centre and Near Core's Edge

Planes	Core Centre Flux (10^{13} neutrons/cm ²)		Near Edge Flux (10^{13} neutrons/cm ²)	
	Group 1	Group 2	Group 1	Group 2
1	3.38	0.723	0.661	0.106
2	9.05	1.94	1.77	0.285
3	14.3	3.07	2.80	0.451
4	19.0	4.07	3.71	0.598
5	22.8	4.89	4.46	0.719
6	25.7	5.51	5.02	0.810
7	27.5	5.89	5.37	0.865
8	28.1	6.02	5.49	0.884
9	27.5	5.89	5.37	0.865
10	25.7	5.51	5.02	0.810
11	22.8	4.89	4.46	0.719
12	19.0	4.07	3.71	0.598
13	14.3	3.07	2.80	0.451
14	9.05	1.94	1.77	0.285
15	3.38	0.723	0.661	0.106

I. The Axial Profiles Are Smooth

Unlike the radial flux profiles, the axial flux profiles are smooth, as shown in Figure 4-4. While the assemblies are arranged in an alternating pattern by enrichment in each plane of the core, the enrichments in the axial direction are uniform. Thus, the assemblies along the core's centre have the same enrichment and thus their fluxes follow the same profile. The same is true for the assemblies near the edge of the core.

II. The Assemblies Have Larger Fluxes in Group 1 Than in Group 2

The fast flux is due to fast neutrons which are produced during fission reactions. When the fast neutrons interact with the nuclei in the core, some of them are absorbed in the resonance region of U-238 and U-235. Thus, the neutrons which do become thermalized constitute a thermal flux which is lower than the fast flux.

III. The Assemblies Along the Core's Centre Have Larger Fluxes Than Assemblies Near the Core's Edge

The assemblies within the core experience less neutron leakage than the assemblies which are nearer the core's edges. Thus, the neutron flux for the axial centre-line of the core is larger than the neutron flux for the axial assemblies near the core's edge.

4.2.4.2 The Power Profiles

The power profiles corresponding to the flux profiles in Section 4.2.4.1 are presented in this section. The power values were calculated using the equation $\text{Power} = V \times E_f \times \Sigma_f \times \phi$ (Lamarsh & Baratta, 2001). The assembly volume is $V = 1.12518 \times 10^4 \text{ cm}^3$ (VEPCO, 2009), and the fission energy for U-235 is $E_f = 200 \text{ MeV}$ (Lamarsh & Baratta, 2001). The values of Σ_f were determined by dividing the $\nu \Sigma_f$ values in Table 4-3 by ν . The group 1 $\nu \Sigma_f$ values were divided by $\nu = 2.349$ while the group 2 values were divided by $\nu = 2.432$ (Stacey, 2001).

The radial power profiles are presented in Table 4-8 while the axial flux profiles are presented in Table 4-9. The radial and axial power profiles are shown in Figure 4-5 and Figure 4-6 respectively. Since the power profiles are obtained by scaling the flux profiles in Figure 4-2 and Figure 4-4, their shapes and properties are identical to those already encountered with the flux profiles. Therefore, the power profiles will not be discussed in this section, but presented instead. Any observation which can be made from the power profiles in this section can be explained by the flux profiles discussion in Section 4.2.4.1.

Table 4-8: Radial Power along the Middle of Planes 1 and 8

Fuel Assemblies	Plane 1 Power (W)		Plane 8 Power (W)	
	Group1	Group2	Group1	Group2
1	7.09E+03	2.30E+04	5.89E+04	1.91E+05
2	1.18E+04	4.44E+04	9.83E+04	3.70E+05
3	2.01E+04	6.92E+04	1.67E+05	5.75E+05
4	2.09E+04	7.87E+04	1.74E+05	6.55E+05
5	2.92E+04	1.01E+05	2.42E+05	8.36E+05
6	2.68E+04	1.01E+05	2.23E+05	8.41E+05
7	3.40E+04	1.17E+05	2.83E+05	9.76E+05
8	2.89E+04	1.09E+05	2.40E+05	9.06E+05
9	3.40E+04	1.17E+05	2.83E+05	9.76E+05
10	2.68E+04	1.01E+05	2.23E+05	8.41E+05
11	2.92E+04	1.01E+05	2.42E+05	8.36E+05
12	2.09E+04	7.87E+04	1.74E+05	6.55E+05
13	2.01E+04	6.92E+04	1.67E+05	5.75E+05
14	1.18E+04	4.44E+04	9.83E+04	3.70E+05
15	7.09E+03	2.30E+04	5.89E+04	1.91E+05

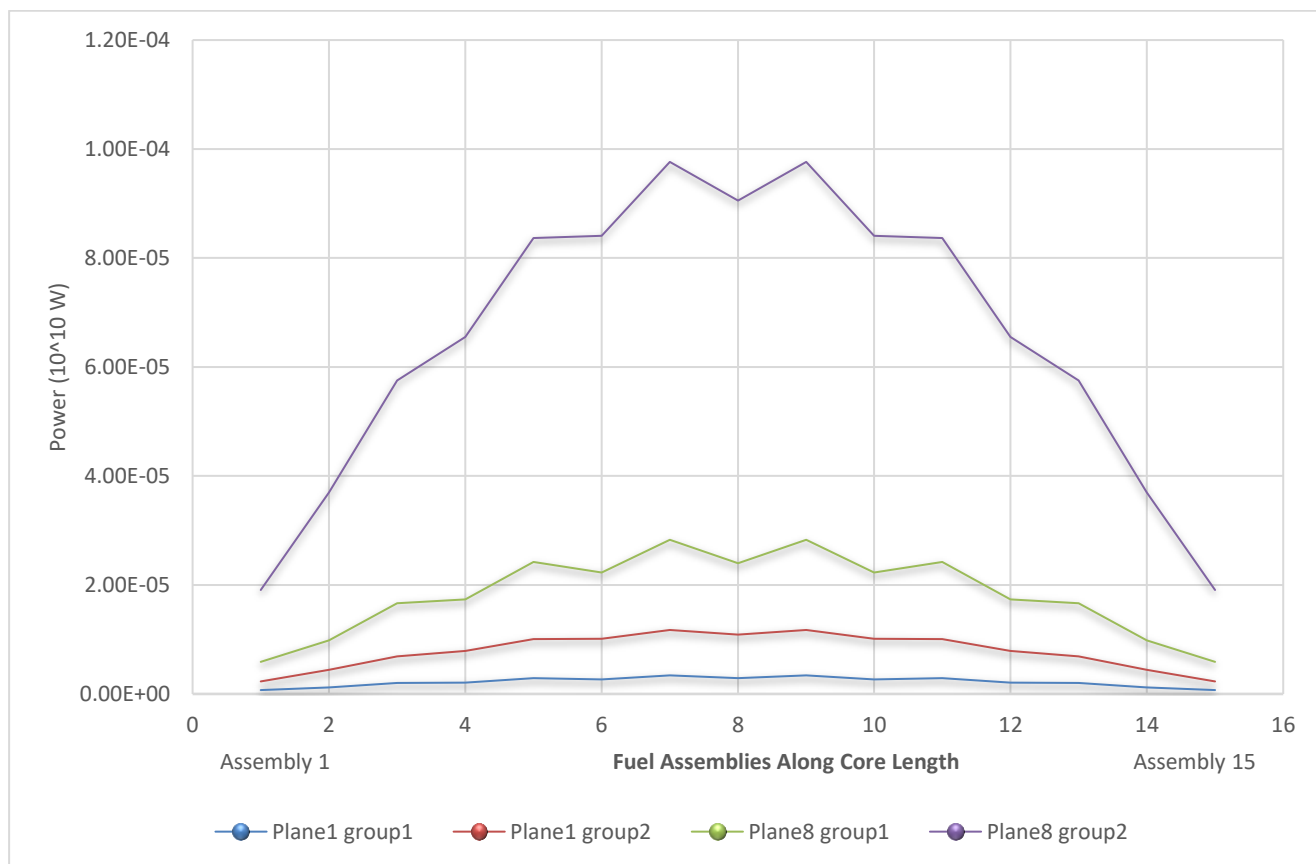


Figure 4-5: The 2-group Radial Power Profiles for Planes 1 and 8

Table 4-9: Axial Power along the Centre and Near Core's Edge

Planes	Core Centre Power		Near Edge Power	
	Group1	Group2	Group1	Group2
1	2.89E+04	1.09E+05	1.18E+04	4.44E+04
2	7.74E+04	2.92E+05	3.17E+04	1.19E+05
3	1.23E+05	4.62E+05	5.02E+04	1.89E+05
4	1.62E+05	6.12E+05	6.65E+04	2.50E+05
5	1.95E+05	7.37E+05	7.99E+04	3.01E+05
6	2.20E+05	8.29E+05	9.00E+04	3.38E+05
7	2.35E+05	8.86E+05	9.62E+04	3.62E+05
8	2.40E+05	9.06E+05	9.83E+04	3.70E+05
9	2.35E+05	8.86E+05	9.62E+04	3.62E+05
10	2.20E+05	8.29E+05	9.00E+04	3.38E+05
11	1.95E+05	7.37E+05	7.99E+04	3.01E+05
12	1.62E+05	6.12E+05	6.65E+04	2.50E+05
13	1.23E+05	4.62E+05	5.02E+04	1.89E+05
14	7.74E+04	2.92E+05	3.17E+04	1.19E+05
15	2.89E+04	1.09E+05	1.18E+04	4.44E+04

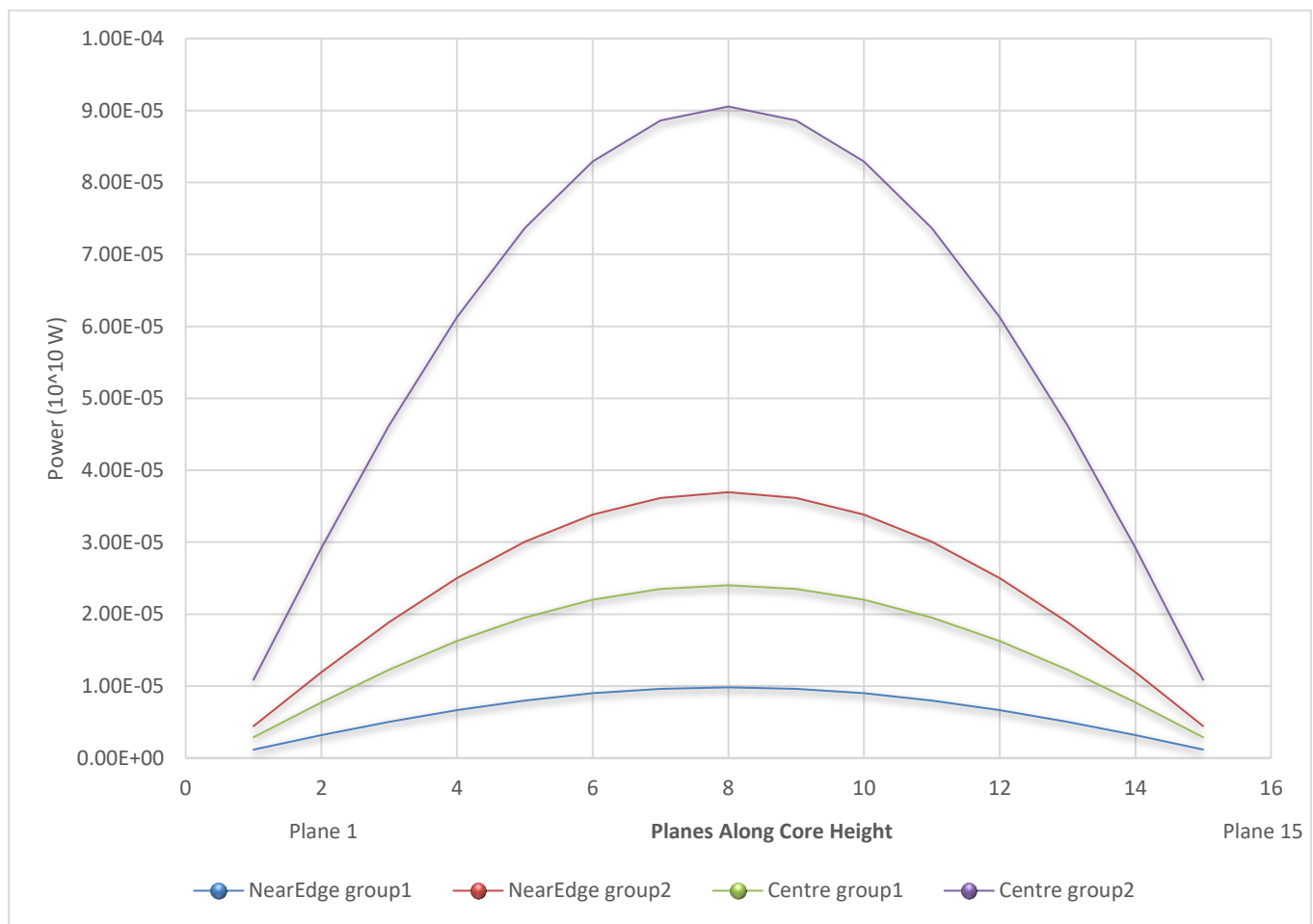


Figure 4-6: The 2-group Axial Power Profiles along the Core's Centre and Edge

CHAPTER 5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

The DRAGON and DONJON source codes were acquired, compiled and linked into DRAGON and DONJON executables. The executables were able to exactly reproduce reference results provided by the DRAGON and DONJON development team

The North Anna fuel assemblies were modelled in DRAGON. The infinite multiplication factors and 2-group constants produced for the models during DRAGON lattice calculations were found to be consistent with expected results from literature.

The North Anna full reactor core was modelled in DONJON. The effective multiplication factor, flux and power profile results produced for the model during full core DONJON calculations were found to agree with expected results according to literature.

As per the research aim, DRAGON and DONJON were tested on the North Anna reactor and they were found to produce plausible results.

Therefore, DRAGON and DONJON can be used for neutronics research projects which involve the types of calculations which were tested in this research work.

5.2 Recommendations

- The DRAGON models should be further developed to include burnable poisons and control rods insertion.
- The DONJON model should be tested when considering the water assemblies around the core.
- DRAGON and DONJON should be further tested on burn-up calculations.
- Spacer grids should be accounted for in future modelling.
- Coupled thermal hydraulics calculations should also be done for the DONJON and DRAGON models.
- The self-shielded cross-sections for the North Anna fuel cells and control rods should be calculated in the future instead of using libraries which are already self-shielded.
- In this research project, the meshes of the DRAGON and DONJON models were assumed to be optimized from the reference models provided by the developers of DRAGON and DONJON. However, mesh studies should be conducted to check for convergence.

BIBLIOGRAPHY

Anon., n.d. *integrated publishing*. [Online]

Available at: <http://nuclearpowertraining.tpub.com/h1019v2/css/Figure-2-Effects-Of-Over-And-Under-Moderation-On-Keff-49.htm>

[Accessed 05 november 2019].

Anon., n.d. *Polytechnique de Montreal*. [Online]

Available at: www.polymatl.ca/phys/en/donjon-download

[Accessed 16 June 2019].

ATI, 2019. *ATI Metals*. [Online]

Available at: www.atimetals.com/products/pages/zircaloy-4.aspx

[Accessed 20 June 2019].

Britannica, 2019. *Britannica*. [Online]

Available at: www.britannica.com/science/law-of-large-numbers

[Accessed 25 June 2019].

Case, K. & Zweifel, P., 1967. *Linear transport theory*. s.l.:s.n.

Choi, H. Y. & Park, C. J., 2017. Conceptual design of thorium based epithermal spectrum reactor. *Annals of nuclear energy*, Volume 109, pp. 61-68.

Dictionary, Y., 2018. *Your Dictionary*. [Online]

Available at: www.yourdictionary.com/neutronic

[Accessed 17 June 2019].

DoE, 2010. *Executive summary of the draft integrated electricity resource plan for South Africa - 2010 to 2030*, s.l.: Department of Energy, South Africa.

DoE, 2018. *Draft Integrated Resource Plan 2018 - For public comments*, s.l.: Department of Energy, South Africa.

DoE, 2019. *Integrated Resource Plan 2019*, s.l.: Department of Energy, South Africa.

Encyclopedia, 2017. *Wikipedia*. [Online]

Available at: https://en.wikipedia.org/wiki/neutron_transport

[Accessed 16 June 2019].

Energy, U. D. o., 1993. *nuclear physics and reactor theory*. Washington, D.C.: Office of scientific and technical information.

Gupta, C. K., 1989. *Materials in nuclear energy applications*. 1st ed. s.l.:CRC Press.

Hebert, A., 2000. *The GAN generalized driver*, Montreal: Polytechnique de Montreal.

Ipsen, I. C., 2009. *numerical matrix analysis:linear systems and least squares..* s.l.:s.n.

Lamarsh, J. R. & Baratta, J. A., 2001. the slab reactor. In: L. Curless, ed. *Introduction to nuclear engineering*. s.l.:Prentice Hall, pp. 271-274.

Lamarsh, J. R. & Barrata, A. J., 2001. *Introduction to nuclear engineering*. 3rd ed. s.l.:Prentice Hall.

Lamarsh, J. R. & Barrata, A. J., 2001. Neutron attenuation. In: L. Curless, ed. *introduction to nuclear engineering*. s.l.:Prentice Hall, pp. 57-60.

Lamarsh, J. R. & Barrata, A. J., 2001. nuclear reactors and nuclear power. In: L. Curless, ed. *introduction to nuclear engineering*. s.l.:Prentice Hall, p. 176.

Lamarsh, J. R. & Barrata, A. J., 2001. temperature effects on reactivity. In: L. Curless, ed. *Introduction to nuclear engineering* . s.l.:Prentice Hall, pp. 365-376.

LANL, 2019. *Los Alamos National Laboratory, MCNP*. [Online]

Available at: <https://mcnp.lanl.gov>

[Accessed 19 June 2019].

Lewis, E. & Miller, W., 1985. *Computational Methods for Neutron Transport*. New York: Wiley and Sons, Inc..

Liou, K. N., 1973. A numerical experiment on Chandrasekhar's discrete ordinates method for radiative transfer: Application to cloudy and hazy atmospheres. *J Atmos sci*, Volume 30, pp. 1303-1326.

Lux, I., 2018. *MC particle transport methods; neutron and photon calculations*. [Online]

Available at:

https://www.researchgate.net/publication/328399316_Monte_Carlo_Particle_Transport_Methods_Neutron_and_Photon_Calculations

[Accessed 26 June 2019].

Marleau, G., 2001. *DRAGON theory manual part 1: Collision probability calculations*, s.l.: Ecole Polytechnique de montreal.

Marleau, G., Hebert, A. & Roy, R., 2014. *A userguide for DRAGON*, s.l.: s.n.

Mosteller, R., 1991. Moteller Test Geometry. *Nuclear Science Engineering*, Volume 107, p. 265.

Nahavandi, N., Minuchehr, A., Zolfaghari, A. & Abbasi, M., 2015. apatially adaptive hp refonement approach for PN neutron transport equation using spectral element method. *Annals of nuclear energy*, Volume 85, pp. 1066-1076.

Naicker, V., Du Toit, M. & Nyalunga, G., 2015. *NWURCS User Guide*, s.l.: North West University.

Nichols, J. P., 1962. *Soluble Neutron Poisons as a Primary Criticality Control in Shielded and Contained Radiochemical Facilities*, s.l.: Oak Ridge National Laboratory.

NJOY, 2016. *NJOY*. [Online]

Available at: www.njoy21.io

[Accessed 29 June 2019].

ORNL, 2019. *Oak Ridge National Laboratory*. [Online]

Available at: <https://ornl.gov/scale>

[Accessed 19 June 2019].

Poltechnique, 2019. *Polytechnique de Montreal*. [Online]

Available at: www.polymatl.ca/phys/en/dragon-download

[Accessed 16 June 2019].

power, n., n.d. *nuclear power*. [Online]

Available at: www.nuclear-power.net/nuclear-power/reactor-physics/neutron-diffusion=theory/neutron-transport-theory-boltzmann-transport-equation/

[Accessed 19 june 2019].

Roberts, A. J., 2010. *Research Gate*. [Online]

Available at: <https://www.researchgate.net/publication/303471187>

[Accessed 16 June 2019].

Stacey, W. M., 2001. Multigroup Diffusion Theory. In: *Nuclear Reactor Physics*. New York: Wiley and Sons, Inc., pp. 123-134.

Stacey, W. M., 2001. neutron-induced nuclear fission. In: *nuclear reactor physics*. New York: John Wiley and Sons, Inc, pp. 3-13.

Stacey, W. W., 2001. neutron transport theory. In: *nuclear reactor physics*. New York: Wiley and Sons, Inc., pp. 295-375.

Stefan, T. T., 1998. Discrete-Ordinates method in radiative heat transfer. *International journal of engineering science*, 36(12-14).

U.S.DOE, 1993. *nuclear physics and reactor theory*. Washington, D.C.: Office for scientific and technical information.

U.S.NRC, 2019. *United States Nuclear Regulatory Commission*. [Online]
Available at: <https://www.nrc.gov/reading-rm/basic-ref/glossary/nuclear-poison-or-neutron-poison.html>
[Accessed 01 November 2019].

U.S.NRC, n.d. *U.S.NRC Library*. [Online]
Available at: www.nrc.gov/reading-rm/basic-ref/glossary/neutron-flux.html
[Accessed 16 June 2019].

Varin, E., Roy, R., Herbet, A. & Koclas, J., 2005. *A User Guide For DONJON*, s.l.: Ecole Polytechnique de Montreal.

VEPCO, 2009. *north anna power station updated final safety analysis report chapter 4*, north anna power station: U.S.NRC.